No	C_Title	C_First Name	C_Last Name	C_Affiliation	C_Country	Paper Title	Abstracts	Topic	Presentation Preference	Reviewers	Review Comments
1 IAEA 26	Mr	Alexander	Sapozhnikov	Federal Environmental, Industrial and Nuclear Supervision Service of Russia 109147 Moscow, Taganskaya, 34, Russia	Russia	New Safety Requirements Addressing Feedback from the Fukushima Daiichi Accident	[illegible]s well known that regulatory body activities are constantly aimed at improving the safety of the supervised nuclear facilities. These [illegible]nerally include amending safety requirements, implementation of licensing and periodic safety reviews (PSRs), and inspections, and this is a 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, especially those, subjected to extreme external events. It was proposed that the research reactor 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. Rostechnadzor 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 complementary 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 was also 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).	IAEA Workshop		Mark	Accepted-Oral
2	Dr	Gilles	Bignan	French Atomic Energy Commission (CEA) – Cadarache Centre 13108 St-Paul-Lez-Durance - France	France	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	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 Gilles Bignan(1), Jean-Yves Blanc(1), Jérôme Estrade (1), Pascal Chaix(2) French Atomic and Alternative Energies Commission (1) Nuclear Energy Division (2) International Relations Division Cadarache and Saclay Research Centres France Corresponding author: gilles.bignan@cea.fr 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, - 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. The paper will present in detail the CEA offer as an ICERR, the template agreement and it will describes, as examples, some first scientific and technical actions recently launched with the Affiliates.	General session		Danas	Accepted -Oral
3	Mr	Stephane	Gaillot	CEA,DEN,DTN,Nuclear Technology Department CEA Cadarache. F-13108 Saint-Paul-lez-Durance,FRANCE	France	JHR Project: Irradiation devices. In-service inspection of nuclear pressure equipment's. Investigation of non destructive examinations techniques for inspection purposes	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 and risky operation. To overcome such problems, the possibility of using non-destructive examination (NDE) tech-niques 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.	New Project	Oral Presentation	Lin-Wen	Accepted-Oral

4	Dr	Yaosong	Shen	Institute of Applied Physics and Computational Mathematics	China	Disposing High-level Transuranic Waste in Subcritical Reactors	We propose a new method of burning high-level transuranic (TRU) waste combinedi 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 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.	Innovative Methods	oral Presentation--> Poster	Rob	Accepted-Oral with comment: clearly define **(Th[U+FF0D]U)**
5	Prof	Ezra	Elias	Technion Israel Institute of Technology	Israel	Analysis of a Hypothetical LOCA in an Open Pool Type Research Reactor	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	Innovative Methods	Oral Presentation	Gilles	Accepted-Oral with comment that the abstract is more developed (half a page)
6	Mrs	Muriel	Antony	Commissariat à l'Energie Atomique et aux Energies Alternatives	France	Moly Production in the Jules Horowitz Reactor: Capacity and status of the development	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(*)/Cadarache in France. JHR will contribute to the security of supply of medical radioisotopes, especially for the 99Mo-99mTc. Four locations are devoted to the 99Mo 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 were 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).	New RR or Radio-Isotopes production	Oral Presentation	Nestor	Accepted-Oral
7	Mr	Emmanuel	Grolleau	IRSN	France	Overview of Ameliorations and Modifications implemented on French Research Reactors since the Fukushima Daiichi accident	After the Fukushima Daiichi accident that occurred in Japan in March 2011, the French Nuclear Safety Authority (ASN) asked to all operators of nuclear facilities to carry out a reassessment of their installation taking into account the lessons learned from the Fukushima accident feedback. Since that time, a lot of discussions and technical exchanges between the operators, the ASN and the Institute of Radiological protection and Nuclear Safety (IRSN) took place concerning the definition of safety reinforcements and modifications to be implemented to enhance the capability of nuclear facilities to withstand extreme events such as earthquake, flooding or tornadoes. In France, such reassessments have not only concerned the nuclear power plant but also fuel cycle facilities and all research reactors currently in operations or under construction. After a summary of main principles and approach retained in France for the definition of technical and organizational provisions to put in place to ensure the safety in case of extreme situations, the paper will present a global overview of ameliorations and modifications that have been implemented (and are now operational) since 2011 on French research reactors and their related nuclear sites. The article will underline the most important topics in terms of safety objectives and concrete realizations on facilities. Finally, the paper will present the modifications that must be finalized to totally complete the consideration of the Fukushima accident feedback.	IAEA Workshop		David	Accepted - Oral
8	Mr	Christian	Gonnier	CEA	France	Experimental devices in Jules Horowitz reactor and first orientations for the experimental programs	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 CEA /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, ADELINE 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).	New project		John	Accepted Oral

9	Mr	Nestor	Delorenzo	INVAP	Argentina	Training of the First Operation Team	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 decide 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.	General session	oral Presentation	Mark	Accepted-Oral
10	Mr	Nestor	Delorenzo	INVAP	Argentina	Research Reactor Design Drivers	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.	New RR	Oral Presentation	Claude	Accepted Oral
11	Mr	Tzach	Makmal	McMaster University	Canada	MTR-type core elements improvements for optimization of radioisotopes production	The primary purpose of research reactors (RRs) is to provide a neutron source for research in natural sciences, industrial processing and nuclear medicine. The latter takes place in 25% (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 and (ii) Irradiation Position body material. The study presents a 3-D MTR fuel calculation using Serpent. 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.	Radio-Isotopes production or Innovative methods		Hoan-Sung	Accepted-Oral
12	Mr	Tzach	Makmal	McMaster University	Canada	Validation of the stable period method against analytic solution	Control rod (CR) reactivity worth plays an important role in safety and control of reactors. The determination of the reactivity worth is essential to assure safe and reliable operation of the reactor system. There are a number of 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 found that there are two main parameters that contribute to the accuracy of the method: Firstly, the effective delayed neutron fraction (beff gb), which is the effective fraction of neutrons that born delayed. The effectiveness weighting factor, g (importance factor), may range between 1 to 1.25 depending on the core size and the enriched fuel level. Variation within the importance factor range can lead up to 26% uncertainty in the reactivity estimation. Secondly, the waiting time that the operator allows the precursor neutrons to die out. The waiting time depends on the value of the reactivity insertion and can change dramatically the quality of the experiment. 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 regulating 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 regulating rod is withdrawn a percentage of it is 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 the absolute deviation between the estimated reactivity values is up to 1.3%. Additionally, it found that the doubling time method is more suitable for incremental regulation rod calibration of more than six sub-segments/ increments.	Innovative Methods		Hoan-Sung	Accepted-Oral
13	Mr	Robert	Schickler	Oregon State University	USA	Installation of a second CLICIT irradiation facility at the Oregon State Triga Reactor	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. Reactivity effects and facility flux values were obtained through k-code calculations and F4 tallies. Theoretical flux values were verified by irradiating Al-Au wires to determine thermal and epithermal spectra.	Utilisation		Alexander	Accepted-Oral
14	Mr	Andrew	Eltobaji	ANSTO	Australia	ANSTO OPAL Reactor CNS Replacement	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.	Utilisation-CNS		Nestor	Accepted-Oral

15	Dr	Xueming	Shi	Institute of Applied Physics and Computational Mathematics	China	Progress in conceptual research on Fusion Fission Hybrid Reactor for energy	Fusion Fission Hybrid Reactor for Energy(FFHR-E), which is fueled by natural uranium and cooled by light water, can accelerate the early application of fusion energy and make fuller use of uranium resources. A simplified conceptual model based on ITER (International Thermal Experimental Reactor) is constructed. In order to model the burnup problems, a three dimensional code MCORGS, which is coupled by MCNP and ORIGENS, is developed. A new point-wise cross section library NuDa-C, which consists of nearly 300 nuclides under multiple temperatures, is made based on ENDF/B-VII.1. A simplified pyro-reprocessing scheme is suggested. It is expected that the spent fuel can be heated up to 2100K by its decay heat, the fission product elements with boiling points below 2100K will be evaporated. Neutron irradiation damage on materials is also evaluated, a reload period around five years is then suggested and 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. Computational Fluid Dynamics code is used in steady state heat transfer analysis. The maximal temperatures in different zones are all lower than their allowable values and show a reasonable margin. The MDNBR of coolant is 5.22, which is bigger than that of PWR's. The coolant pressure drop in fission zone is 39.6Kpa, amounts to 0.255% of inlet pressure.	New RR projects	oral Presentation	Steven	Accepted -oral with comments: the scope of the paper is too broad to cover all quoted aspects from the abstract in a high quality way: please reduce the topics of the
16	Mr	Claude	Pascal	TechnicAtome	France	Contributions of previous projects to the design of new research reactors	For the successful achievement of a new research reactor design, it is necessary to meet on one hand the Customer specification and on the other hand Safety Authorities requirements. Customer needs are often centered on a common set of applications, however their balance is tailored to each specific project, hence it results 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: [U+F02D] as regards utilization and operation performances and the good project achievement [U+F02D] as regards potential consequences for the operator, 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 constantly pushing the designer to have the ability to challenge his solution 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 could be implemented in the design of new RRs. The topics of concern are: [U+F02D] the reactor overall architecture as regards utilization and operation performances [U+F02D] the experimental devices ensuring experimental and production applications [U+F02D] the SSCs ensuring utilization, operation or safety functions as regards their proven design characteristics and qualification requirements [U+F02D] 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 to current projects and are illustrated with some examples in the light of AREVA and TechnicAtome experience.	New RR projects		Nestor	Accepted-Oral
17	Mr	Maciej	Lipka	National Centre for Nuclear Research	Poland	Possible shifts in MARIA reactor reactivity and power changes caused by the seismic event	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.	Utilisation-Safety		Ayman	Accepted-Poster
18	Dr	Rian	Prinsloo	NECSA	South Africa	Recent developments of the OSCAR calculational system, as applied to selected examples from IAEA research reactor benchmarks	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 (loosely 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 analysis. 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. This deployment to various codes 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. Such consistent models are highly desirable, since holistic reactor calculational support requires a suite of codes for various applications. In addition to defining a unified model for all	Innovative Methods		Khalifeh	Accepted-Poster
20	Prof	Kwon-yeong	Lee	Handong Global University	South Korea	Investigation on core downward flow by a Passive Residual Heat Removal System of Research Reactor	Most of existing research reactors have been designed with Active Residual Heat Removal System (ARHRS) to remove decay heat which is continually generated after the primary cooling pump stops. However, ARHRS takes much cost and is hard to design. Therefore, an investigation is carried out to verify a design of Passive Residual Heat Removal System (PRHRS) of research reactor.[1] PRHRS mainly consists of three parts; a flywheel linked to the primary cooling pump maintains core downward flow even though the pump stops; Gravity Core Cooling Tank (GCCT) makes Core Downward Flow continuously by a differential head between reactor pool and GCCT; and flap valves change the direction of the core flow to upward. In this research, the performance of GCCT is analyzed in terms of hydrodynamics, specifically mass flow rate, with analytical and experimental methods. An experimental facility for investigating PRHRS of research reactor is designed and manufactured in small scale. Furthermore, a theoretical model based on Bernoulli equation and CFD model are developed to predict performance of GCCT for the experimental facility scale. To conduct the experiment in different experimental conditions, two sizes of Differential Pressure Pipe(DPP) are selected as a variable. 0.75 and 1 inch DPP are given for GCCT test. The analytical (theoretical and CFD) results are compared based on the experimental results. The highest mass flow rate of experimental results are much small than that of two models. It is because the pressure loss of experimental facility is bigger than that of two models. Considering this reason of difference, theoretical and CFD model are improved to follow the experimental results by adjusting the pressure loss of DPP. To apply these	Safety or innovative Methods		Claude	Accepted - Poster

21	Prof	Kwon-yeong	Lee	Handong Global University	South Korea	Investigation of siphon breaker simulation program through small scale siphon breaker experiment	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 severe accidents. To prevent it, siphon breaker is developed. However, as it is difficult to predict the results, a siphon breaker simulation program (SBSP) was designed Lee and Kim[1]. In this study, by using the SBSP, a small scales siphon breaker was designed to verify the SBSP. Range of experiments included general range, Kang et al.[2], 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. Reference: (1) K. Y. Lee, and W. S. Kim, "Development of siphon breaker simulation program of investing loss of coolant accident of a research reactor", Annals of Nuclear Energy, Vol. 101, pp.49-57 (2017) (2) S. H. Kang, H. S. Ahn, J. M. Kim, H. M. Joo, K. Y. Lee, K. Seo, D. Y. Chi, J. Yoon, G. D. Jeun and M. H. Kim, "Experimental study of siphon breaking phenomenon in the real-scaled research reactor pool", Nuclear Engineering and Design, Vol. 255, pp. 28-37 (2013).	Innovative Methods		Christopher	Accepted-Poster
22	Prof	Kwon-yeong	Lee	Handong Global University	South Korea	Study of an integrated passive safety system for a research reactor	An innovative integrated passive safety system for a research reactor is proposed in this study to improve the safety of the research reactor. This integrated system has three functions in the facility as a decay tank, siphon breaker, and long-term cooling tank. This paper also deals with the process of designing and optimizing the decay tank and the siphon breaker of the integrated passive safety system using CFD. The decay tank was conservatively designed with the minimum residence time of 60 s. Using the DPM function, it was found that the flow residence time is at least 67 s and more than 70% of the particles, specially N-16, have a residence time between 60 and 100 s. In addition, the performance of a new type of siphon breaker installed in the proposed decay tank model was tested. We confirmed that siphoning occurred when there was a pipe rupture accident in the research reactor, and the reactor core became exposed to air without the siphon breaker. Therefore, we designed an 18-inch-diameter siphon breaker at the top of the decay tank, and we could observe the breaking of the siphon that prevented the occurrence of a severe accident in the research reactor. We could also use the coolant of the decay tank for long-term cooling of the research reactor. In conclusion, this new integrated safety system provides the three functions of a decay tank, siphon breaker, and long-term cooling tank respectively. Therefore, the available space can be efficiently used with a single structure. As this structure of the facility is simple, its construction cost is low and it is easy to maintain; hence, it will be easier to secure the safety of research reactors using this design.	new projects or Safety		Mark	Accepted-Oral
23	Mr	Vincent	Roux	CEA	France	Findings and results of safety reassessments and safety improvements on the ORPHEE research reactor	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 hasards 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 Orphee 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 sismicity risk. The safety improvements mainly consisted in the implementation of reactor emergency shutdown on seismic signal, implemented to strenghten 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 decribe the relevant findings from the periodic safety reassessment and post-Fukushima stress tests, and safety improvements that were implemented on Orphee reactor to fulfill the requirements of safety levels from the regulatory body.	IAEA Workshop		John	Accepted-Oral
24	Mr	Mohamed	Khalifa	TechnicAtome	France	Hydraulic design and validated calculation tool of the Jules Horowitz Reactor (JHR) reflector	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, thermalhydraulic 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.	Innovative Methods		Ayman	Accepted-Oral

25	Dr	Lin-wen	Hu	MIT Nuclear Reactor Laboratory	USA	Preliminary results of in-core irradiation tests of fluoride salt and materials at the MIT research reactor	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 hightemperature 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 R 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 (t1/2 = 7.1 s,) and 19O (t1/2 = 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.	Utilisation- new concept		Kashima-Gilles	Accepted Oral
26	Dr	Khalifeh	AbuSaleem	JAEC	Jordan	Commissioning of the Jordan Research and Training Reactor (JRTR)	The JRTR is a multipurpose reactor designed and constructed to be used for education and training, research and radioisotope production. It has been recently commissioned where it went critical using an external neutron source in April 2016. The purpose of commissioning for research reactors 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 standards on commissioning of research reactors have been followed. As recommended in the Safety Guide NS-G-4.1, the JRTR commissioning process was divided into three main stages with hold points as appropriate. These stages are; tests prior to fuel loading, fuel loading tests and initial criticality tests which include low power tests; and the last stage was dedicated to power ascension tests and power tests up to rated full power. During the commissioning process, several tests have been carried out including Construction Acceptance Tests (CAT), System Performance Tests (SPT), Integrated System Tests (IST) and Reactor Performance Tests (RPT). The tests confirmed that all design and performance parameters have been achieved. Particularly, the design specifications on nominal thermal power, thermal neutron flux and negative reactivity feedback have been met. Currently, the JRTR is in the operational mode. This paper describes in detail each commissioning stage of the JRTR and the final results and conclusions.	New RR		Herman	Accepted-Oral
27	Mr	Jin	Lu	CIAE	China	Safety Analysis for prototype MNSR HEU core unloading and storage (full paper)	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 ϕ241×278mm 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 isϕ280×309mm, outer dimension is ϕ540×691mm, 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.	Safety		David	Accepted - Oral.
28		Jianlong	Li	CIAE	China	The Cold Neutron Source is a key experimental facility of CARR Reactor in China	The Cold Neutron Source（CNS）is a key experiment facility of 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/cm^{2}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 thermohydraulic performance have been tested and verified. It has been demonstrated that all aspects of the thermohydraulic design feature have been fulfilled.	CNS		Rob	Accepted-Oral-Very important milestone for CARR

29		Matias	Marticorena	CNEA	Argentina	On-line condition monitoring tool for nuclear research reactors coolant system components	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 behavior 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.	Utilisation		Gilles	Accepted-Oral
30		Jarunee	Kraikaew	Office of Atoms for Peace, National Nuclear Forensics Laboratory	Thailand	Developing of nuclear material database for nuclear security encouragement	The National Nuclear Forensics Laboratory (NNFL) of Thailand was established in 2013 by Office of Atoms for Peace (OAP) under the Project No. 30, "Network of Excellence for Nuclear Forensics in South East Asia Region(2013-2014)", supported by EU CBRN CoE. Its principle objective is to enhance the capability of the nation to overcome illicit trafficking of domestic radioactive and nuclear material. The National Nuclear and Radiological Emergency Plan was issued and enforced to response for nuclear and radiological accidents/incidents/threats and any terrorism. These attempts support nuclear security regime following Safeguards Agreement (SG) and Non-Proliferation Treaty (NPT). At present, OAP conventional radioactive and nuclear materials database is applied to support safety regulation of all radiation facilities in Thailand. Nuclear materials in only location outside facilities were developed as Microsoft Access 2010 Database in 2016. In order to support National Nuclear Forensics Database, the said database will be developed follow the structures and formats available for Nuclear Forensics Library, relating to IAEA concepts. The hypothetical entries will be setup from the fields appropriate to all categories of nuclear and other radioactive materials, including general and technical information. The success of this operation will reduce timely and informed nuclear assessments using this developing database, as well as deterring concerned illicit activities. All missions will assist the National Nuclear and Radiological Emergency Plan, as well as the country nuclear security and physical protection related to SG for the peaceful use of atomic energy.	Security		Tim	Accepted-Poster
31	Dr	Khalifeh	AbuSaleem	JAEC	Jordan	Enhancement of the safety fo the Jordan Research and Training Reactor (JRTR)	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 Fukoshima-Daicci 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 h 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 behavior 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	IAEA Workshop		David	Accepted - Oral. This should be included in the IAEA Workshop session. Please correct spelling errors and typos. Clarification needed on the IAEA safety standards applied.
32		Sudi	Ariyanto	BATAN	Indonesia	Competence development of research reactors personnels in Indonesia	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 reources 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 retension program can be done, and research reactors can be operated in a safe, secure and sustainable. Planning for the development of competency of research reac will be developed follow the structures and formats available for Nuclear Forensics Library, relating to IAEA concepts. The hypothetical entries will be setup from the fields appropriate to all categories of nuclear and other radioactive materials, including general and technical information. The success of this operation will reduce timely and informed nuclear ass	General session		Danas	Accepted-oral with comments:Please correct spelling errors Clarification needed: -what method has been used for self-evaluation: IAEA standard ?-
33		Sudi	Ariyanto	BATAN	Indonesia	Public Education and outreach for supporting nuclear program in Indonesia	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 generati 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 behavior 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. 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	General session		Christopher	Accepted-oral
34	Mr	Steven	Van Dyck	CEN	Belgium	The new irradiation infrastructure at the BR2 reactor	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	Utilisation or Refurbishment		Mark	Accepted-Oral

35 (IAEA 8)	Dr	Katarzyna	Niedzwiedz	Federal Office for the Safety of Nuclear Waste Management	Germany	Safety reassessment of German research reactors in the light of the accident at the Fukushima Daiichi nuclear power plant - current status of the improvements	The first actions following the accident at the Fukushima Daiichi Nuclear Power Plant on 11th March 2011 were taken in Germany already a few days later. On 14th March 2011 the federal government and the competent prime ministers of the federal states requested a comprehensive plant specific safety reassessment - stress test - of nuclear power plants. On 7th of July 2011, just after the stress test for nuclear power plants has been conducted, a similar comprehensive safety reassessment was requested also for all German research reactors with a continuous thermal power of more than 50 kW. The general approach for the safety reassessment of research reactors was based on the stress test for nuclear power plants. However, due to the comparable lower radioactivity inventory and the lower risk potential of research reactors, the assessments criteria had to be adjusted individually. The corresponding catalogue of requirements and the assessment of the robustness of research reactor facilities has been carried out by 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 behavior 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. 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 mol	IAEA Workshop		David	Accepted - Oral.
36	Dr	Lin-wen	Hu	MIT Nuclear Reactor Laboratory	USA	Progress conversion study of the MIT research reactor from highly enriched uranium to low enriched uranium fuel	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.	Fuel		Gilles	Accepted-Oral
37	Mr	Djalal	Hamed	Nuclear Research Center of Draria	Algeria	Numerical solution of transient natural convention in vertical heated rectangular channel between two vertical parallel MTR-type fuel plates	The aim of this paper is to perform a numerical solution by finite volume method of transient natural convection in a narrow rectangular channel between two vertical parallel MTR-type fuel plates, imposed under a heat flux with a cosine shape. To determine the margin of the nuclear core power at which the natural convection mode can ensure a safe core cooling, where the cladding temperature should not reach the specific safety limits (90C). For this purpose a computer program is developed to determine the principal parameter related to the nuclear core safety such as the temperature distribution in the fuel plate and in the coolant (light water) as a function of the reactor core power. Our results are validate throughout a comparison against the results of the work done by D. Jo & all [1], which is considered like a reference of this study.	Innovative Methods		Christopher	Accepted-oral
38	Prof	Daeseong	Jo	Kyungpook National University	South Korea	Onset of flow instability in a rectangular channel under transversely uniform and non-uniform heating	Flow instability in a narrow rectangular channel (2.35 mm x 50.0 mm x 250 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 behaviors in a narrow rectangular channel under transversely uniform and non-uniform heating.	Innovative Methods		Christopher	Accepted-oral
39	Mr	Nir	Hazensprung	SNRC	Israel	Ageing management and structures, systems, and components improvements at IRR1	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.	Utilisation		Gilles	Accepted-Oral
40	Mr	Maurice	Mashau	NECSA	South Africa	Application of the next generation of the OSCAR code system to the ETRR-2 multi-cycle depletion benchmark	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 development in the OSCAR code system focused on integrating high-fidelity and standard nodal diffusion analysis methods in a consistent way. A code independent front end system can be used to create a detailed heterogeneous model of a reactor. This single model can then be used to create input for various underlying codes including MCNP, Serpent and the OSCAR nodal diffusion solver (named MGRAC), ensuring consistency between the models for different codes. In this work, a detailed heterogeneous model is built in OSCAR from which a reference Serpent model is generated. Additionally, cross section and model input are also generated for use in the MGRAC. Several control rod calibration experiments are simulated with both Serpent and MGRAC, 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,	Innovative Methods		Claude	Accepted-oral

41	Mr	Simon	Nicolas	TechnicAtome	France	Monte-Carlo coupled depletion codes efficiency for research reactor design	Early stages of core design and industrial studies require a quick and efficient calculation of key neutronic parameters (reactivity, control rods efficiency, power peak factors, core material balance, etc.) at any given step during core cycle. This determination is mainly achieved by deterministic calculation schemes and TechnicAtome has developed its own tool named COCONEUT (COre COnception NEUtronic Tool) [1] [2] dedicated to research reactor calculation. The aim of this tool - based on deterministic codes APOLLO2 [3] [4] (2D, multigroup transport theory) and CRONOS2 [5] (3D, diffusion theory) – is to be generic and to perform accurate calculations of MTR-type reactors with limited CPU-time. 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 (TRIPOLI4© [6], MCNP6 [7] and Serpent [8]) and the deterministic scheme COCONEUT. This comparison also provides the Validation and Verification process (V&V) undergone by COCONEUT [9]. This is carried out on standard fuel assembly (FA) and control FA of a core with burnable poison and used by TechnicAtome to set new methodology studies. Parameters such as reactivity, isotopes concentration of interest and neutronic flux are studied on a burn-up calculation reaching 150 GWd/tU. Code specifications concerning the implementation of the burnup calculation and time calculation are also important with the purpose of choosing a reference code for industrial studies.	new projects		Khalifeh	Accepted-oral
42	Mr	Simon	Nicolas	TechnicAtome	France	A dummy core for V&V and education and training purposes at TechnicAtome: in and ex-core calculations	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 an all-purpose dummy core. This non-existing core provides a mixture of most common features found in research reactors throughout the world. Its characteristics enable us to validate and qualify both calculational 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 order to enable objective comparisons in methodologies, TechnicAtome has designed an all-purpose dummy core. This non-existing core provides a mixture of most common features found in research reactors throughout the world. Its characteristics enable us to validate and qualify both calculational 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. 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. Finally, the paper illustrates the sketch of TRIPOLI4© and GEANT4 calculation with unification at geometry level and show very preliminary neutronic results based on GEANT4 with further outlooks.	Innovative Methods		Nestor	Accepted-Poster
43	Mr	Gilbert	Rouviere	CEA	France	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.	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 characterisation (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. Conclusion JHR, the new high performance MHR under construction in France has taken into account the lessons learned from Fukushima Daiishi accident. After stress test evaluation, a Hardened Core of material dispositions was decided by ASN based on a CEA proposal, constituted by already designed SSC and new SSC. New methodologies have been defined to guarantee Hardened Core SSC operation in post Fukushima situation preventing a severe accident or limiting its progression, limiting large-scale releases in the event of an accident which is not possible to control, enabling the licensee to perform its emergency management duties. The completion of Hardened Core implementation on JHR will be performed without start up schedule modification.	IAEA Workshop		David	Accepted - Oral
45	Mr	Jason	Chakovski	ANSTO	Australia	Improvements in the safe availability and reliability of the OPAL Reactor	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.	utilisation		Alexander	Accepted-oral
46	Mr	John	Bus	ANSTO	Australia	Radiation safety training at the Open Pool Australian Light-water (OPAL) multi-purpose reactor	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,	Education & Training		Xinxin-Gilles	Accepted-Oral
47	Mr	Claude	Pascal	CEA	France	RCC-MRx 2015 Code: Context, overview and on-going developments	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 feedbacks 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.	New projects		Ayman	Accepted-oral

48	Mr	Florian	Nicolas	IRSN	France	Handling safety of French Research Reactors	The French Research Reactors (RRs) currently in operation or in a shutdown state have been commissioned in the sixties or in the seventies. The safety analysis carried out at their design stage mainly focused on preventing core accidents. Handling-related risks were taken into account but not thoroughly analysed. Handling safety is yet an important issue on RRs as operations like fuel supply, radioactive waste disposal and preparation of experiments require the manipulation of heavy loads. Moreover, these operations usually take place nearby the reactor and storage pools, even during reactor operation, in relatively small-sized areas due to the usually modest size of RRs. In the past decades, handling safety has become a major issue for operators who are now expected to present a comprehensive safety analysis of these operations. The French Nuclear Safety and Radiation Protection Institute (IRSN), as the technical support organisation to the French Nuclear Safety Authority (ASN), is facing an increasing number of relating cases. The use of containers in RRs to manipulate fuel, spent fuel, waste and other radioactive material from experiments and irradiations stands for a large number of handling operations. Current international standards for the transport of radioactive material impose the use of more solid, resistant and heavy container than those taken into account when RRs were designed. These containers are manipulated inside the nuclear installations and require an appropriate design of the handling equipment and of the civil engineering of the building. For instance, in order to evacuate the spent fuel of the Phebus reactor in Cadarache, south-east of France, its operator, the Alternative Energies and Atomic Energy Commission (CEA), had to develop inventive technical solutions. In the same way, the Laue Langevin Institute (ILL), operator of the High Flux Reactor (RHF) in Grenoble, east of France, had to deal with a similar difficulty to evacuate operational waste. The French regulation requires operators to carry out a Periodic Safety Review (PSR) of their facility which implies an evaluation of the compliance of the reactor with the current standards and the state-of-the-art. In this regard, the CEA conducted in 2004 the latest PSR for Cabri, a reactor located in Cadarache. Conclusions have highlighted the need to improve the safety of heavy loads manipulation. Cabri and RHF reactors are due to conduct a PSR end of 2017. Handling safety is one of the major topics that will be assessed by IRSN. On the basis of the previous examples, this paper will present an outlook of IRSN reviews on handling topics, which includes identification of improvements, operational and safety constraints as well as solutions found by operators.	Safety		David	Accepted - Oral
49	Mr	Alfio	Arcidiacono	ANSTO	Australia	Cold Neutron Source Helium Injection Logic Modification at OPAL	The Cold Neutron Source (CNS), located at ANSTO's OPAL Reactor, utilises a helium cryogenic refrigeration loop to keep deuterium in liquid state for the production of cold neutrons. The CNS in-pile structure is located within the vacuum containment vessel which has two primary functions: provide a vacuum environment to aid in thermal insulation of the CNS in-pile, and to prevent damage to the surrounding reactor sub-systems in the event of CNS failure. 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 thimble. The original protection logic primarily focused on prevention of hot damage. However, a once-off undesired injection of helium into the vacuum containment brought cold damage into focus. Although this event did not appear to cause damage to the CNS in-pile structure, an administrative control has been placed to prevent reoccurrence. A thorough analytical analysis on both hot and cold damage was performed by ANSTO which led to the software logic regarding helium injection to be redesigned. This paper discusses the mechanisms of hot and cold damage to the CNS in-pile and the subsequent software logic modifications, ultimately leading to the new arrangement whereby the automatic injection of helium only occurs if the temperature of the in-pile was above 0 C prior to a cryogenic helium circuit trip. This provides an engineering solution and removes the associated administrative controls. A provision to manually inject blanketing helium is still available but not envisaged to be used in standard operation.	CNS		Tim	Accepted as oral
50	Prof	Oscar	Peire	Universidad Nacional de Rosario	Argentina	The nuclear technology and it's development in the National University of Rosario - Argentina	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 technicias 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 Univerity 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 Nuclears Studies and Ionizating Radiations)dedicated to the development of the subject. It works with a staff of professionals 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 programed experiences, laboratory work and pre – profesional 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 wih 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 presentatios in congresses and symposiums.	Education & Training		Kashima-Gilles	Accepted-oral
51	Dr	Weijian	Lu	ANSTO	Australia	OPAL Cold Neutron Source Moderator Performance	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.	CNS		Rob	Accepted-Oral
52	Dr	Masaji	Arai	Japan Atomic Energy Agency	Japan	Thermal-hydraulic conceptual design of the new multipurpose research reactor succeeding to JRR-3	The new multipurpose research reactor is a 30 MW thermal power, swimming pool type research reactor and will be constructed instead of JRR-3 for utilization of the neutron beam, irradiation, training and so on. Light water will be used as coolant and moderator. The reactor core will be surrounding by heavy water. The reactor will use low enriched U-Mo plate-type fuel. The core thermal-hydraulic conceptual design of the new reactor 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 heat flux is less than 283 W/cm2 for the forced convection cooling mode at a core power of 30 MW with the pressures of 1.68 kg/cm2 at the core inlet and coolant velocity in the standard fuel element of about 5.5 m/s. The results obtained in this work establish the preliminary technical specifications for the core thermal-hydraulic design of the new multipurpose research reactor.	New projects		Herman	Accepted-Oral

53	Prof	Ayman	Hawari	North Carolina State University	USA	Multidisciplinary engagement at research reactors: The NCSU PULSTAR	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	Utilisation		Heiko	Accepted-oral
54	Mr	Zhenping	Chen	University of South China	China	Modelling and simulation of dispersion particle fuels in Monte Carlo neutron transport calculation	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 Thorium Molten Salt Reactor and High Temperature Gas Cooled Reactor. However, the dispersion fuel element in which the fuel particles statistically distributed in the background material presents new challenges for the conventional reactor physics methods. Based on the high-fidelity Monte Carlo method, the various modeling and simulation methods of dispersion particle fuels in neutron transport calculation was studied. The basic principle and implementation strategis of the Sub-Fine Lattice Method (SFLM), Random Sequential Packing Method (RSPM) and Chord Length Sampling Method (CLSM) were presented. And the impacts on the modeling efficiency and calculation accuracy of the methods were given. The numerical results showed that modeling methods can preliminarily meet the requirements of the Monte Carlo neutron transport calculations for the dispersion particle fuel.	fuel		Tim	Accepted as oral
55	Mr	Mario	Carta	ENEA	Italy	TAPIRO fast spectrum research reactor characteristics for neutron radiation damage analyses	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.	Utilisation		Steven	Accepted-oral
56	Mr	Mario	Palomba	ENEA	Italy	Testing newly developed thermal neutron gem detectors at the ENEA TRIGA research reactor	The thermal neutron radial channel of ENEA TRIGA RC-1 reactor was selected as a suitable field of test for evaluating the performance of new detectors, specially developed for thermal neutron monitoring in material science, medical and basic research applications. The field offered at ENEA TRIGA is especially suited for the tests because of the following characteristics: the irradiation area is located in a closed hutch with sufficient internal area, the beam can be controlled by the user through an automated shutter, the thermal fluence rate is very stable and easily tunable from few tens of n*cm-2*s-1 up to approx. 1*106 n*cm-2*s-1 and the gamma contamination of the field is moderate. This report describes the results of irradiation tests performed on two types of GEM (Gas Electron Multipliers) detectors. The first one has a side-on configuration with 16 borated silicon sheets and a window of 1 x 9 cm2. The second one has been realized in an head-on configuration with four borated aluminum grids and an active area of 10 x 10 cm2. The principal aim of the tests at TRIGA was the neutron efficiency estimate. This is particularly important for the side-on detector which has been conceived as valid substitute of the 3He-tubes in neutron diffraction studies.	Utilisation		Khalifeh	Accepted-poster
57	Dr	Daniel	Parrat	CEA	France	Networking advanced experimental capacities in operating European materials testing reactors for qualification of innovative nuclear fuel and materials: The FIJHOP R&D program proposal	Ageing of operating service oriented Material Testing Reactors (MTRs) in Europe lead to significant changes (real or potential) in the irradiation infrastructure landscape (e.g. OSIRIS definite shutdown end of 2015). This trend leads globally to a decrease in available experimental capacity and skilled resources. On the other hand, and given the time-frame for developing and qualifying new nuclear fuels and materials, both sustainable and state-of-the-art research capacity are required in existing MTRs for support to the nuclear power industry. To counteract the risk to face insufficient adequate European experimental capacities to fulfill middle- and long-term requests from industry and Safety Authorities, two main ways are currently considered as efficient: • The Jules Horowitz Reactor (JHR), under construction at CEA Cadarache and foreseen to be in operation by the beginning of the next decade, will reinforce the link between the operating European MTRs (BR2, HFR, LVR-15, HBWR, MARIA. . .) to perform advanced experiments. Aim is to promote their coupled operation, as an irradiation service network, • The implementation of multilateral programs associating one or several MTRs and hot cell laboratories for post-irradiation examinations (PIEs). These programs beneficiate from a strong support of state-of-art models and codes, either for defining the experimental protocol fulfilling the request, or pre-calculating the sample behaviour or finally assessing results. This allows in turn enlarging databases and validation domain of participant's simulation tools, even under development. 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. This proposal of multilateral	new project		Lin-Wen	Accepted-oral

58	Prof	Ken	Nakajima	Kyoto University	Japan	Present status of Kyoto University research reactor, KUR	By reflecting the lessons learned from the accident of TEPCO's Fukushima-Daiichi Nuclear Power Plant which occurred in March 11th 2011, the Nuclear Regulation Authority (NRA) has formulated the new law regulating the nuclear facilities including the research reactors. Then, all the research reactors in Japan had to temporary shut down, 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 March, 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. Presently, we have almost finished the refurbishment works and the inspections by NRA, and we plan to start KUR operation in August. In this report, we will describe the NRA's new regulation for research reactors and our response to it for the case of KUR.	IAEA Workshop		Xinxin-Gilles	Accepted-oral
59	Mr	Mark	Summerfield	ANSTO	Australia	Some thoughts on operator intervention arising from safety reassessments of research reactors in the light of the Fukushima Daiichi NPP accident	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.	IAEA Workshop		Steven	Accepted-oral-very relevant content
60	Dr	Heiko	Gerstenberg	Technical University of Munich	Germany	Production of medical isotopes at the FRM II research reactor	The most modern German reserach reactor FRM II is operated on the campus of the Technical University of Munich. Due to its heavy water moderated compact HEU core it is ideally suited for neutron beam tube research. Nonetheless FRM II offers several irradiation channels exhibiting high neutron flux densities combined with a very pure thermal neutron spectrum. These features are attracting a growing interest for the production of radioisotopes in particular for radiopharmaceutical purposes. As of today Lu-177 is the most important product in this sense. It is activated indirectly exposing Yb-oxide as starting material to a high thermal neutron fluence of typical 1.5E20 1/cm². The activation product is transferred to itg, a commercial company on site FRM II for further processing to carrier free Lu-177 n.c.a. Another radiosiotope of increasing importance is Ho-166 to be used in form of microspheres for the radioembolization of hepatic malignancies. For this therapy the sound condition of the organic microspheres encapsulating the Ho-166 is essential. Due to its pure thermal neutron spectrum and low temperature in the irradiation environment FRM II is ideally suited to meet these requirements. This work is done in collaboration with the Dutch company Quirem medical. Last not least FRM II applied at the relevant authorities for a license to install and operate an irradiation facility aiming the production of Mo-99/Tc-99m as a fission product. The concept of the facility and its integration into the FRM II reactor pool will be presented along with the status of the project.	Radio-Isotopes production		Danas	Accepted-oral
61		Wade	Marcum	Oregon State University	USA	Observations on experimental fluid structure interactions of plate-type fuel	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 modeling 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.	fuel		Hoan-Sung	Accepted-oral
62		Roland	Ruiterman	NRG	Netherlands	The impact of changes in utilization on human performance	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.	utilisation		John	Accepted-oral

63	Mr	Nick	Howarth	ANSTO	Australia	Secure enterprise integrations for multipurpose research reactors	The OPAL research reactor operated by the Australian Nuclear Science and Technology Organisation is a multi-purpose scientific and manufacturing facility. Information produced by the reactor's control system is relied upon by of scientific and manufacturing information systems, including: Neutron beam line data acquisition systems; and ANSTO's Enterprise Resource Planning (ERP) system for the scheduling silicon Neutron Transmutation Doping (NTD) irradiations, and the irradiation of materials for analysis and the production of radiopharmaceuticals. Each consumer of information produced by the OPAL research reactor has its own computer security model ranging from the collaborative network operated by the Australian Centre for Neutron Scattering (ACNS), to the secure network requirements of the ANSTO ERP platform. Through the application of international guidance from the IAEA, national frameworks and procedures from the Australian Signals Directorate (ASD), and local expertise, ANSTO has developed and implemented a security architecture that meets the requirement of all users of the OPAL research reactor while protecting the key control systems that ensure safe, secure, and sustainable operation. This paper provides an overview of the development and implementation of this security architecture and provides suggestions and lessons learned that may be of use to other similar facilities.	Security		Heiko	Accepted-oral fits perfectly into the scope of IGORR
64	Mr	Sebastien	Gay	CEA	France	Jules Horowitz reactor: RCC-MRX applicability for the design phase of experimental devices	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*, 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 equipments, 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. There 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. This paper describes the applicability of the RCC-MRx* rules within the instructions for use, the input data, the general analysis for mechanical calculations and the criteria verifications. This process is illustrated by applications focused on experimental devices in development. *Design and Rules for Mechanical Components of Nuclear Facilities	New projects		Herman	Accepted-oral
65	Dr	Anton	Kastenmuller	Technische Unviversitat Munchen	Germany	First regular PSR of the FRM II	The FRM II is the most modern Research reactor in Germany and got its first criticality in March 2004. 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. After the nuclear startup phase the FRM II has begun routine operation end of April 2005. According to the German nuclear energy act and a dedicated requirement from the approval of operation 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 will present the grade approach for the PSR for a research reactor in respect to the mandatory German regulations for NPPs, some results of the PSR and status of the follow up process underway to finalize the PSR.	Safety		Kashima-Gilles	Accepted -oral
66	Prof	Myung Hyun	Kim	Kyung Hee University	Korea	Role of AGN-201K for university education in Korea	AGN-201K in Kyung Hee University (KHU) is a zero power reactor. This reactor has been the only open facility for training and education in Korea since 2008. Reactor Research & Education Center (RREC) is an administration organization for student training. There are six experimental modules provided for a hands-on experimental short course. Domestic programs inviting undergraduate students are usually done for five days as dormitory-housing programs. In-house education for KHU was done 8 times for 118 students. 35 training courses were done for 360 students from 8 universities. Recently 12 hands on experimental courses were provided 12 times to Japanese, Emirates and Jordanian students. RREC of KHU has been working with Kindai University for the international nuclear experiment school since 2014. Now students from 5 Japanese Universities (Kindai, Kyoto, Kyushu, Nagoya and Fukui) and KHU are operating three short courses; 2 in Japan and one in Korea. As domestic and international demands for hands on experiment, RREC is now facing the limitation in acceptable number of participants throughout the year. Internet reactor laboratory (IRL) is assigned to KHU as a host reactor for Asia-Pacific region. This is now planned and prepared for the development of both hardware and software equipment. Also class protocol should be modified to be effective in on-line broadcasting. Because of cyber security measure, data communication between a host site and many guest sites is not allowed for all aspects. Nevertheless IRL may open new opportunity to support universities in other Asian countries with expanded capacity in participating student number.	Utilisation		Khalifeh	Accepted-oral
67	Mr	Byunghun	Hwang	KAERI	Korea	Completion of seismic rehabilitation project at HANARO after the Fukushima Daiichi accident	This is to report the completion of HANARO reactor building reinforcement project that began in 2015 as part of the HANARO safety reinforcing endeavors. Right after the Fukushima accident in 2011, KAERI immediately carried out necessary safety reconfirmation program in accordance with the NSSC (Nuclear Safety and Security Commission) of the government's safety reassurance policy. As a result of the safety inspection, the NSSC requested KAERI to reassess the seismic qualification of the HANARO with particular emphasis on the reactor building and the stack. The NSSC special safety inspection team assessed whether this initial standard can also withstand large scale earthquakes such as the one at Fukushima. For the seismic assessment of all reactor structures, the EPRI-NP-6041-SL of the Methodology for Assessment of Nuclear Power Plant Seismic Margin (Rev. 1) has been applied. The outcomes of the assessment proved that the seismic margin of the reactor and the reactor concrete island that accommodate reactor structure and major reactor systems was Richter scale 7.7 which was more than qualified. However, it was found out that some area of the outer wall of the reactor building does not satisfy the seismic design criteria. The findings were reported to the NSSC in December 2014 and the Commission requested KAERI in March 2015 to reinforce those areas identified as dissatisfaction. To this effect, KAERI immediately began the HANARO outer wall reinforcement project. Fig. 1 shows the reinforcement concept. Steel H-beams were used to support the reactor building both from inside and outside. It was designed that the H-beams will share seismic impact that can be applied on outer wall of the reactor building. It was confirmed that the reinforcement concept will be effective based on an in-depth analysis as well as a real scale test. Through the seismic reinforcement, both the axial and flexural strengths were greatly improved. Meanwhile, PS tendon was used for some part of the outer wall of the reactor building in order to improve specific seismic performance. As of May 2017, all works have been completed including installation of the built-up sections. KAERI will re-operate the HANARO with the regulatory authority's approval and also explain to citizen's verification team that is organized by the local government. With the completion of the HANARO safety reinforcement and rehabilitation project, KAERI hopes that it will demonstrate its effort to secure safety for the country and for the people.	IAEA Workshop		Xinxin-Gilles	Accepted-oral
68	Mr	Jin-Won	Shin	KAERI	Korea	Status of periodic safety review of HANARO	The first Periodic Safety Review (PSR) for HANARO (Hi-flux Advanced Neutron Application ReactOr) 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 has 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.	Safety		Alexander	Accepted-oral

69	Dr	Itzchak	Dahan	NRCN	Israel	Radiation resistance of the U(Al, Si)3 alloy: ion-induced disordering	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.	Innovative Methods		Kashima-Gilles	Accepted-Poster
70	Dr	Ewald	Liebhart	Mirion Technologies	USA	Upgrade of aging neutron flux monitoring systems in research reactors on the example of implementations in TRIGA reactors	One of the most common research reactor designs is GA's TRIGA® reactor of which more than 30 are currently operational worldwide. As this is one of the oldest research reactor designs many facilities face the issue of obsoleteness for vital components of their reactor control and protection systems. Mirion Technologies' proTK™ offers a full range of digital neutron flux monitoring channels which operate for 25 years in nuclear power plants and research reactors around the world. Due to the modularity of these channels, they can be configured to fit into most existing nuclear facilities with no or very little adaption necessary to existing interfaces on site. Although a complete upgrade of the I&C in one single step has many advantages, the cost of such a full-scale project may pose a problem to many research and training facilities. It has been proven in practice that due to the modularity of proTK neutron flux monitoring channels - and consequently the variety of possible configurations - it is achievable to perform a system-by-system upgrade with a minimal impact on the availability of the facility. An overview of available proTK neutron flux monitoring channels that are suitable for upgrading corresponding TRIGA systems will begiven. The upgrade of the wide range neutron flux monitoring system with the digital wide range channel DWK 250 will be presented in more detail, as this involves the implementation of more specific functions. Finally, cyber security aspects related to the use of digital signal processing will be discussed, including features that are implemented in proTK channels to ensure safe operation.	O&M		Danas	Accepted-Poster
71	Mr	Pablo	Ramirez	CNEA	Argentina	Management of safety and licensing requirements during the RA-10 reactor construction stage	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. Thorugh 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. 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.	New project		Alexander	Accepted-poster
72	Mr	Pablo	Cantero	CNEA	Argentina	The IAEA internet reactor laboratory project: status, feedback from recent broadcasting and future expansion	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.	Utilisation		Ayman	Accepted-oral
73	Mr	Jinsen	Xie	University of South China	China	Theoretical study of steady state neutron flux reconstruction in ADS subcritical reactor by using higher order modes	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.	Innovative Methods		Lin-Wen	Accepted-Poster
74	Dr	Victor	Gillette	University of Sharjah	UAE	Attending remote research reactor experiments: an evaluation from University of Sharjah	In 2012, the University of Sharjah introduced the Department of Nuclear Engineering to the undergraduate Diplomas it offers to Emirati and international students, mainly from the Gulf countries. The graduates of this program are expected to satisfy, in the long term, the demand of human resources needed in the Barakah Power Reactors, at least partially. A PhD degree in Nuclear Engineering can also be obtained in the UAE. Senior students have to take a course of Advanced Laboratory of Nuclear Engineering, which consists of performing several experiments in a Research Reactor. These experiments are followed remotely while run at the University of Massachusetts at Lowell. In this presentation, the experience of 4 batches of students will be discussed. The intended long-term strategy to be followed in successive semesters will also be discussed.	Utilisation		John	Accepted-oral very important topic

75	Mr	Herman	Blaumann	CNEA	Argentina	The construction stage in the RA-10 reactor project	The RA-10 is a new multipurpose research reactor which 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 on 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 6th. 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. Commitment with safety culture was reinforced by assuming that, beside the contractors participation, the responsibility for implementing the licensee rely 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.	new project		Alexander	Accepted-oral
76	Ms	Kathryn	McCarthy	CNL	Canada	Filling the neutron gap at the Canadian Nuclear Laboratories after shutdown of the National Research Universal (NRU) Reactor	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, and is considering multiple options to fill the neutron gap. This paper discusses the options under consideration and progress towards identification of the best way to fill the neutron gap at CNL.	General session		Gilles	Accepted-oral
77	Mr	Pierre	Gavoille	CEA	France	Post irradiation testing capabilities of experimental reactor components at the LECI facility for service life assessment	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.	Utilisation		Gilles	Accepted-oral
78	Mr	Mohammad	Kassim	Malaysian Nuclear Agency	Malaysia	35yrs experience in Operations and Utilization of the Malaysian PUSPATI TRIGA reactor	The Malaysian PUSPATI TRIGA Reactor (RTP) reached its first criticality on 28^{th} 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.	Utilisation		Gilles	Accepted-oral

79	Mr	Kyoungwoo	Seo	KEARI	South Korea	Commissioning Experience for Reactor and Primary Cooling System of Jordan Research and Training Reactor (JRTR)	The primary cooling system is designed to cool the heat generated from the core of the 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 maintained clean cautiously during commissioning because re-cleaning or disassembling and re-assembling will require additional time and cost. Thus, reactor, fluid equipment, instrument and pipes should be fabricated by the cleaning procedure in accordance with the requirements of the related code and standard. The reactor and primary cooling system should be installed by the installation procedure because the interface between the reactor package and the related system including fluid system, platform, pool door, instrument, detector conduit, and pool covers inside the pool is considerably complicated. After the primary cooling system is installed to reactor and pool, the system flushing is performed to remove the 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 to measure the system flow rate and pressure loss and check the function of 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 the in-service inspection test program can be performed after the results of SPTs satisfy the acceptance criteria of tests.	new projects		Gilles	Accepted-oral
80	Mr	Jean-Sebastien	Zampa	CEA	France	Safety Reassessment of OSIRIS Reactor in the light of Fukushima Daiichi Accident	A safety reassessment was carried out on the OSIRIS reactor, immediately following the Fukushima Daiichi accident. The margins were evaluated for severe external events such as earthquakes or flooding. The additional effects of the loss of electrical supply and of the ultimate heat sink were also analyzed. On the basis of these evaluations improvements were chosen to reinforce potential weak points and avoid cliff-edge effects. These improvements were technical and organizational, including procedures, human resources, emergency response organization or use of external resources. Most modifications were implemented during the years 2013 and 2014. A further periodic safety reassessment is under preparation. It will include an analysis of the lessons learnt from past operations, conformity analyses and new safety studies to comply with new safety standards and new French regulations and requirements. The reassessed safety demonstration will also take into consideration the diminution of nuclear risks after OSIRIS reactor was definitively shut down at the end of the year 2015 and will include those related to the preparation of dismantlement activities.	IAEA Workshop		Gilles	Accepted-oral
81	Mr	Ram Charan	Sharma	IAEA	Austria	IAEA activities in support of operation and maintenance of research reactors	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.	O&M-Utilisation		Gilles	Accepted-oral
82	Dr	Khalid	Almarri	The British University	UAE	A qualitative study for establishing the conditions for the successful implementation of public private partnerships in research reactor project in newcomer countries	The UAE is currently developing a peaceful nuclear energy program. Research of nuclear energy technologies is required to support nuclear energy generation projects and maximize their performance. Research of this type will require building an 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 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 experts in the field of research reactors, using grounded theory method. Causal conditions that stemmed from project initiation work were found to be the main barriers for the success of PPPs in RR. This included the reduction of government intervention to help mitigate the risk of failure. 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.	General session		Gilles	Accepted-oral ;important topic
83	Mr	Mohamed	Al Jaberi	The British University	UAE	A study of the impact of cultural diversity on the technological innovation process in the nuclear energy corporations	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 Emirates Nuclear Energy Corporations, which has been chosen for conducting the primary research. In the present context 75 employees of the Emirates Energy Corporation (ENEC) 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.	General session		Gilles	Accepted-oral

84		Silva	Kalcheva	SCK-CE, BR2 Reactor	Belgium	Feasibility studies for simultaneous irradiation of NBSR & MITR fuel elements in channels H3 & H5 fo the BR2 reactor	The BR2 reactor along with other MTR reactors (ATR) 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: the Missouri University Research Reactor (MURR), the Massachusetts Institute of Technology Reactor (MITR), the National Bureau of Standards Reactor (NBSR) and the High Flux Isotope Reactor (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, respectively at the conditions of MURR, MITR, NBSR, HFIR. The first important studies for the feasibility irradiations of all DDE's are the neutronics studies of power and burn-up profile evolution, which are the basis for further thermo-hydraulics studies. The present paper is focused 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 BR2 reactor has a long history examination LTA's of a different reactor, such as the Jules Horowitz Reactor Fuel Elements, which have been successfully tested during 5 years irradiation campaign. The BR2 reactor also has a high flexibility to operate with a small compact core (20-25 fuel elements) up to extended (30-35 fuel elements) with a flexibility to locate the control rods in different channels. In the present paper, the MITR and the NBSR fuel elements are loaded in the 200 mm diameter channels, in the channel H5 and H3 of the BR2 reactor, respectively. The requests for the maximum surface heat fluxes and fission density in the peak regions of the DDE-MITR and DDE-NBSR differ significantly: 64 W/cm2 maximum heat flux in the hot spot of DDE-MITR at BOL vs. 160 W/cm2 in the hot spot of DDE-NBSR at BOL. In order to 'mimic' the irradiation conditions for each DDE in the BR2 reactor as in the original reactors and to reach the mentioned above irradiation targets, a number of optimization scenarios are considered and described in this paper, which include: (i) optimization of the axial positions of both DDE's in the chosen channels; (ii) modification of the DDE-NBSR axial dimensions; (iii) azimuth orientation of the DDE-MITR; (iv) optimization of the content in the surroundings channels, especially important for the DDE-MITR; (v) loading of baskets with absorber material around DDE-MITR; (vi) choice of material for the irradiation baskets. Detailed 3-D power and burn-up time evolution profiles are evaluated with MCNP6. The number of BR2 irradiation cycles for the DDE-NBSR is estimated to be about ~10-11, and for the DDE-MITR – about 27-28 assuming irradiation at similar surrounding conditions during all cycles. These results are valid for the original requests of irradiation targets, 5.8E21 fiss./cm2 at EOL for DDE-MITR and 7.9E21 fiss./cm2 for DDE-NBSR at EOL. Various options with optimization of the surrounding conditions and adapting the BR2 core load configuration show that the number of cycles for DDE-MITR can be reduced by about 30%	fuel		Gilles	Accepted-oral
85		Alexander	Tuzov	ROSATOM	Russia	RIAR AS IAEA ICERR: PILOT TECHNICAL COOPERATION PROJECTS AND FUTURE PROSPECTS	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	General session		Gilles	Accepted-oral
86	Mr	Cheol Hyun	Kang	KAERI	South Korea	Radioactive Radon Effect of Spent Fuel Storage Pool Kr-85 Monitor	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 filed 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, behavior 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.	Utilisation		Gilles	Accepted-Poster
87	Dr	Sean	O'Kelly	Idaho National Laboratory	USA	The first 50 years of operation of the ATR at the Idaho National Laboratory	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.	General session		Gilles	Accepted-oral
88		Robert	Williams	NIST	USA	The NBSR: Celebrating 50 years of neutron research	Fifty years ago, this week, December 7, 1967, the National Bureau of Standards Reactor, the NBSR, was made critical for the first time. The bureau had just completed the move to its new campus in Gaithersburg, Maryland from its cramped laboratories in Washington, DC. The reactor was one of three state of the art neutron sources to start up at about the same time; the HFBR at Brookhaven and the HFIR at Oak Ridge were the other two. The NBSR was initially licensed to operate at 10 MW by the Atomic Energy Commission, and then relicensed in 1984 by the Nuclear Regulatory Commission with a power increase to 20 MW. It was again relicensed in 2009 for 20 more years. The reactor was designed with a very large thimble, 55 cm ID, to accommodate a D2O cold neutron source, installed in 1987, and the success of which led to the construction of the guide hall and eventually to the establishment of the NIST Center for Neutron Research, NCNR. The facility has grown tremendously in the years since the first cold neutrons were directed to the guide hall in 1990. A liquid hydrogen cold source replaced the D2O source in 1995 with a six-fold gain in flux, and within a few years there were about 15 instruments on the 7 original guides. A second expansion project was launched in 2007 that has added a new guide hall with 5 more guides, and a second LH2 cold source in one of the thermal beam tubes for the Multi-Axis Crystal Spectrometer. Through it all, the NBSR has provided beams of thermal and cold neutrons for research in materials science, fundamental physics and nuclear chemistry. NCNR has become a world-class research facility operating 28 instruments, 22 of them using cold neutrons. A brief history of the NBSR, a description of the NCNR facility, and prospects for the future will be presented. An excellent account of the growth of the NCNR from the early days to the present was written by Jack Rush and Ron Cappelletti and can be found at: https://www.ncnr.nist.gov/NCNRHistory_Rush_Cappelletti.pdf	General session		Gilles	Accepted Oral
89		Soo-Youl	OH	KAERI	KOREA	Completion of Jordan Research and Training Reactor Construction Project	In June 2017, the Consortium of Korea Atomic Energy Research Institute (KAERI) and Daewoo E&C had 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 endeavor since the project launched in August 2010.	New RR		Gilles	Accepted Oral

90		A.G.	Eshcherkin	ROSATOM	Russia	POWER RAMPING AND CYCLING TESTING OF VVER FUEL RODS IN THE MIR REACTOR	A series of experiments has been conducted at the MIR reactor to experimentally confirm the maximum inner pressure and stress limits for the water-cooled reactor fuel rod claddings both under normal operating conditions (power ramp) and under power cycling mode. 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 tests. For this purpose, specific techniques have been developed and successfully applied as well as experimental rigs and in-pile gagesDuring this period of time, about 120 VVER fuel rods have been tested under the power ramping and step-up modes (see Figure below); about 20 VVER fuel rods have been tested under the power cycling conditions. In the Figure, marked with transparent marker are fuel rods tested in 1990-2000; marked with colors are fuel rods tested from 2012 till now. These are fuel rods with increased fuel amount, Gd-containing fuel rods and 70MW*day/kgU burnup fuel rods. Marked with solid marker are fuel rods that became leaky during the experiments. The paper presents information about techniques applied at the MIR reactor as well as some tests and post-irradiation examinations results.	Utilization-MTR		Gilles	Accepted-oral
91		A.V.	Alekseev	ROSATOM	Russia	EXPERIMENTAL STUDY OF THE VVER-1000 FUEL RODS BEHAVIOR UNDER THE DESIGN-BASIS RIA AND LOCA IN THE MIR REACTOR	*Since 2001, RIAR has been carrying out experiments in the MIR reactor to obtain data on the VVER-1000 fuel rods behavior under the loss-of-coolant accident (LOCA) and reactivity initiated accident (RIA). Each experiment included the examination of the fuel rods conditions, fuel-to-cladding interaction and fission gas release from irradiated fuel. Several experiments were done under each task with the use of VVER-1000 fuel rods spent at an NPP up to a fuel burnup of 40…70 MW×day/kgU. Post-irradiation examinations were performed in the hot cells. At the first stage of the LOCA experiments it was of priority to define the nature of cladding strain of fuel rods in the FA with a non-uniform heat rate over the FA radius, to define the conditions for the core cooling retaining, to study the degradation of cladding material typical for the effect of the second and third LOCA stages. The object of examination was a VVER-1000 FA consisting of 19 shortened fuel rods, including three ones with spent fuel. By now, two experiments have been finished; the maximal claddingAt the first stage of the LOCA experiments it was of priority to define the nature of cladding strain of fuel rods in the FA with a non-uniform heat rate over the FA radius, to define the conditions for the core cooling retaining, to study the degradation of cladding material typical for the effect of the second and third LOCA stages. The object of examination was a VVER-1000 FA consisting of 19 shortened fuel rods, including three ones with spent fuel. By now, two experiments have been finished; the maximal cladding temperature made up 820 and 940°C.At the second stage, the main task was to define the fuel rod leakage parameters and to study the fuel meat behavior (fragmentation, fuel release out of cladding) in case of cladding rupture. The object of examination was a single fuel rod that made it easier to specify its testing conditions. The maximal cladding temperature achieved during the experiment made up 810°C. At a temperature of 770…780°C and pressure drop of 5.0MPa, the fuel rod lost its leak tightness. During the other experiment, the fuel rod remained leak tight at a temperature of 750°C and pressure drop 5.8MPa. The paper presents the results of experiments and post-irradiation examinations. The RIA experiments were aimed at studying the behavior of 40…60MW·day/kgU burnup fuel rods under the effect of neutron flux pulse. The pulse parameters corresponded to the calculated ones for VVER-1000: amplitude - - 3…3,5; half-width - 2…3s. Since the MIR reactor is operated at a constant power, a physical principle was purposely developed as well as equipment to arrange the pulse mode in a single channel. During the above-mentioned experiments, the initial linear load made up 250W/cm (fuel enthalpy was 2.5×105 J/kg) and the enthalpy excess in a pulse achieved 1.7×105 J/kg. The RIA experiments were performed to study the fission gas release from spent fuel. Since the leakage-causing threshold values of the maximal fuel enthalpy averaged over the radius (~ 5.8×105 J/kg) are much higher as those achieved in the MIR reactor, then no significant changes in the cladding and fuel meat state were expected. The paper presents the results of experiments and post-irradiation examinations.*	Utilisation-MTR		Gilles	Accepted-oral
92		A.L.	Izhutov	ROSATOM	Russia	current and prospective tests in reactor mir	Regarding physical features, reactor MIR is a thermal heterogeneous reactor with a moderator and reflector made of metal beryllium. Regarding design features, it is a channel-type reactor immersed in water pool. Such design allowed an advantageous combination of pool-type and channel-type reactor features. At present, MIR reactor is equipped with the following experimental facilities and devices: - loop facilities that are the experimental base of the reactor and provide for its attractive capabilities;-- hot cells with related facilities; test devices where heat is removed from a tested item by primary circuit water of pool cooling water (used to test fuel and components of research reactors);- critical assembly that is a reactor physical model;- stand to inspect fuel assemblies and fuel rods in the cooling pool A wide range of experimental equipment and parameters allow the following tests and experiment to be carried out:- loop and in-pile tests of VVER fuel rod characteristics under the condition simulating standard and abnormal ones and design-basis accidents; investigation of FGR from leaky fuel rods;- tests of fuel rods and fuel assemblies of propulsion, low-power and floating reactors;- tests of fuel rods and fuel assemblies of high-temperature gas-cooled reactors; - tests of fuel rods and structural materials under simulated PWR conditions, including	General session		Gilles	Accepted-Oral