Fast Reactors and Related Fuel Cycles: Next Generation Nuclear Systems for Sustainable Development FR17

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FAST REACTORS AND RELATED FUEL CYCLES: NEXT GENERATION NUCLEAR SYSTEMS FOR SUSTAINABLE DEVELOPMENT (FR17)

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FOREWORD

It is generally recognized that long term development of nuclear power as part of the world's future energy mix will require fast reactor technology with a closed fuel cycle. Fast reactors in the closed fuel cycle represent both the cornerstone of and a bridge to more sustainable nuclear energy production. In the past few decades, fast reactors have been brought to a high level of maturity by the design, construction and operation of experimental and prototype reactors. A number of countries are actively developing fast reactors and there are several demonstration projects, of a variety of sizes, under study or under construction. Progress has been made as well in the development of related fuel cycles with processes being demonstrated at pilot plant scale.

The IAEA has been supporting the development and deployment of fast reactor technology and serving interested Member States for almost five decades. The Technical Working Group on Fast Reactors (TWG-FR) and the Technical Working Group on Nuclear Fuel Cycle Options and Spent Fuel Management (TWG-NFCO) comprise groups of experts providing advice and support for the implementation of IAEA programmatic activities, reflecting a global network of excellence and expertise in the areas of advanced technologies and R&D. Among the wide range of activities and initiatives carried out under the aegis of these two working groups, the International Conference on Fast Reactors and Related Fuel Cycles is a major event.

The first International Conference on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09) was held in Kyoto, hosted by the Government of Japan. The event provided an appropriate forum to achieve the main objectives of sharing knowledge and exchanging information, experience and innovative ideas among the more than 500 experts from 20 countries and 3 international organizations in attendance. The IAEA organized the second conference, on the theme of Safe Technologies and Sustainable Scenarios (FR13), in Paris, in 2013, hosted by the Government of France. This second event was attended by almost 600 experts from 27 countries and 4 international organizations representing different fields of fast reactor and related fuel cycle technologies. Continuing this effort, in 2017 the IAEA organized the third international conference, on the theme of Next Generation Nuclear Systems for Sustainable Development (FR17). Held in Yekaterinburg and hosted by the Government of the Russian Federation, the FR17 conference was attended by some550 participants from 27 countries and 6 international organizations.

The purpose of the FR17 conference was to provide a forum to exchange information on national and international programmes, and more generally on new developments and experience, in the field of fast reactors and related fuel cycle technologies. By providing a scientific platform for experienced scientists, engineers, government officials, safety officers and fast reactor managers to share their perspectives, the FR17 conference also facilitated the exchange of knowledge between generations. Such exchange can help in choosing the correct path of research to meet the upcoming challenges in the development of fast reactors and related fuel cycles.

As an output of the successful organization of the FR17 conference, these Proceedings provide a summary of the different technical, plenary and young generation event sessions, as well as the full text of the opening, closing and plenary speeches delivered during the conference.

The IAEA would like to express its appreciation to the Government of the Russian Federation for hosting the conference through the State Atomic Energy Corporation “Rosatom”, and to the members of the International Advisory Committee, the International Scientific Programme Committee, the Local Organizational Committee and the Secretariat of the Conference for the commitment shown in organizing and convening the FR17 conference.

The IAEA officers responsible for this publication were V. Kriventsev of the Division of Nuclear Power and A. González-Espartero of the Division of Nuclear Fuel Cycle and Waste Technology.

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1. SUMMARY

1.1. INTRODUCTION

The International Atomic Energy Agency (IAEA) once again brought together the fast reactor and related fuel cycle community by organizing the International Conference on Fast Reactors and Related Fuel Cycles: *Next Generation Nuclear Systems for Sustainable Development* (FR17). This conference was held in Yekaterinburg, the Russian Federation from 26-29 June 2017. The Russian Federation's State Atomic Energy Corporation "ROSATOM" offered to host the IAEA event. One of the main reasons for selecting this venue was that the sodium cooled fast reactor BN-800 was commissioned for commercial operation in 2016 at the Beloyarsk Nuclear Power Plant (NPP), which is located in the vicinity of Yekaterinburg. BN-800 is a successor of the BN-600 reactor that has been in operation at the Beloyarsk NPP since 1980.

The first International Conference on Fast Reactors and Related Fuel Cycles: *Challenges and Opportunities* (FR09) was held in Kyoto, Japan, in 2009. The second IAEA Conference (FR13) was held in Paris, France, in 2013 with the theme '*Safe Technologies and Sustainable Scenarios*'.

The importance of fast reactors and related fuel cycles in ensuring the long-term sustainability of nuclear power has been largely recognized for a long time by the nuclear community. Fast reactors are innovative nuclear reactors that offer several key advantages over traditional thermal reactors in terms of sustainability and radioactive waste management. Operating in a fully closed fuel cycle, fast reactors have the potential to extract 60-70 times more energy from uranium than existing thermal reactors and contribute to a significant reduction in the burden of radioactive waste.

At present, many countries are actively developing reactor, coolant, fuel and fuel cycle technologies. Fast reactor technologies under development include sodium, lead, gas, molten salt and supercritical water cooled systems, as well as hybrids, such as accelerator driven systems. Several demonstration projects, ranging from small to large scale, are under study, design and construction.

For energy systems based on fast reactor to become viable for industrial deployment in the coming decades, designers will have to increase their level of safety in order to gain public acceptance. Harmonization of safety standards at an international level will play a leading role in achieving these goals.

1.2. SUMMARY OF THE CONFERENCE

The purpose of the conference was to provide a forum to exchange information on national and international programmes, and more generally new developments and experience, in the field of fast reactors and related fuel cycle technologies. The first goal was to identify and discuss strategical and technical options that may have been proposed by individual countries or companies. The second goal was to promote the development of fast reactors and related fuel cycle technologies in a safe, proliferation resistant and cost-effective manner. The third goal was to identify gaps and key issues that need to be addressed in relation to the industrial deployment of fast reactors and related fuel cycles. The fourth goal was to engage young scientists and engineers in this field, particularly contributing to the development of innovative fast reactor concepts.

The conference was structured to cover all the major technical aspects of fast reactors and related fuel cycles. The objective of the conference was to provide a forum for the exchange of information on fast reactor and fuel cycle technology advances, and related safety, sustainability, economic and proliferation resistance issues. Another objective was to identify gaps and key issues that need to be addressed towards the introduction of industrial scale fast reactors, and the necessary strategies towards public acceptance. Several existing fast reactors, current construction projects and innovative fast reactor concepts that are under development at national and international levels were reviewed and discussed. The conference started with an opening session and was followed by a plenary session in which national and international programmes on fast reactors and related fuel cycles were presented and discussed. Two more plenary sessions were held every morning, during the next two days of the conference. Major advances in several key areas of technological development were presented during the conference at 47 technical sessions, by approximately 200 oral presentations in eight parallel technical tracks. The titles of the tracks are provided below:

- Track 1: Innovation Fast Reactor Designs;
- Track 2: Fast Reactor Operation and Decommissioning;
- Track 3: Fast Reactor Safety Design;
- Track 4: Fuel Cycle Sustainability, Environmental Considerations and Waste Management;
- Track 5: Fast Reactor Materials (Fuels and Structures) And Technology;
- Track 6: Fast Reactors, Experiments, Modelling and Simulations;
- Track 7: Fast Reactors and Fuel Cycles: Economics, Deployment and Proliferation Issues; and
- Track 8: Professional Development and Knowledge Management.

Additionally, about 200 posters complemented the overall picture of the scientific and the state-of-the-art technical developments worldwide. It is worth noting that all technical papers submitted to FR17 conference were peer-reviewed by the members of the International Scientific Programme Committee (ISPC) and then revised accordingly by the authors.

The conference also included two-panel events devoted to safety design criteria for sodium cooled fast reactors and small and medium-sized fast reactors. A young generation event dedicated to young professionals involved in fast reactor programmes and projects was also organized as a plenary session.

1.3. OBJECTIVES AND STRUCTURE OF THE PROCEEDINGS

The Proceedings have been designed to provide an output of the successful FR17 conference. It is expected to serve as a valuable source of information for specialists and newcomers involved in fast reactor technology and related fuel cycles studies and developments. This publication contains the summary of the conference, major findings, challenges and conclusions resulting from the technical sessions organized within the eight parallel tracks of the conference. In addition, opening session, executive summary, keynote speeches, summaries of the technical sessions and panel sessions, a summary of the young generation event and closing session are included. The individual papers and presentations can be found on the CD-ROM attached to this Proceedings and through the Id number hyperlinks in the online version.

2. OPENING SESSION

2.1. ROSATOM

Opening speech as provided, verbatim.

Alexey Likhachev

Director General, ROSATOM, Moscow, Russian Federation

Dear Ladies and Gentlemen, dear colleagues,

Welcome to the largest international IAEA conference on fast reactors and related fuel cycles.

The future of the global nuclear industry and the closed nuclear fuel cycle development, crucial part of which are fast reactor technologies, are intrinsically linked.

This implies that, in the near future, the world atomic industry will become a truly renewable source of energy based on the principle of radiation equivalency of the energy produced from the source material.

It is no coincidence that the largest conference in history devoted to this topic is being held here, in Russian Federation.

ROSATOM is known as one of the world's leaders in fast reactor technologies. The BN-600 reactor has been in operation at the Beloyarsk NPP for 30 years. Last year, the BN-800 reactor was commissioned. As of today, we have developed the BN-1200 project, the first commercial fast reactor, which is currently undergoing its technical and economic feasibility assessment.

We continue our work on the "Breakthrough" project.

We also work on the development of the first Multipurpose Fast Research Reactor (MBIR) which will lay the basis for an international research Centre. The MBIR will provide the atomic industry with modern and technologically advanced infrastructure for the next 50 years.

I believe that the conference will host fruitful discussions and produce relevant steps towards further development of safe and effective energy technologies regarding fast reactors.

I wish all the conference participants interesting and productive discussions and a positive experience of professional communication.

2.2. INTERNATIONAL ATOMIC ENERGY AGENCY

Opening speech as provided, verbatim.

Yukiya Amano

Director General, International Atomic Energy Agency (IAEA), Vienna, Austria

Ladies and Gentlemen,

I am very pleased to address this third *International Conference on Fast Reactors and Related Fuel Cycles.* I am grateful to ROSATOM for hosting this important event.

I had the pleasure of addressing the two previous conferences, in Kyoto in 2009 and in Paris in 2013.

When I spoke in Paris, the 2011 Fukushima Daiichi accident was still very fresh in our minds. The IAEA was still actively involved in helping Japan respond to the emergency.

Since then, all countries with nuclear power have taken vigorous steps to reassess all aspects of safety and to make necessary improvements.

Global interest in nuclear power continues to grow. In fact, installed nuclear capacity is now the highest that it has ever been at 392 gigawatts electrical.

Twenty new reactors were connected to the grid in the last two years, the highest number since the 1980s. There are now 449 power reactors in operation in 30 countries. Sixty more are being built around the world.

We are seeing two interesting developments. *First,* the centre of expansion in nuclear power has shifted from Europe and North America to Asia. *Second*, developing countries are embarking on nuclear power.

This should not really come as a surprise. Populous countries such as China and India need huge amounts of electricity and also want to reduce greenhouse gas emissions. Nuclear power helps with both.

Likewise, developing countries, especially in Africa, desperately need electricity if they are to achieve sustainable development.

It is appropriate that this conference should be subtitled *Next Generation Nuclear Systems for Sustainable Development.* The IAEA is helping developing countries to use nuclear science and technology to achieve the Sustainable Development Goals.

IAEA projections, which are based on information from Member States, indicate continued growth in nuclear power in the coming decades. But it remains to be seen whether that growth will be modest or significant.

Ladies and Gentlemen,

Fast neutron reactor systems, operated in a fully closed fuel cycle, have the potential to significantly increase the sustainability of nuclear power.

They can extract 60 to 70 times more energy from uranium than existing thermal reactors and reduce the volume and toxicity of the final waste. They will be both safer and more efficient than current reactors.

Fast reactors are flexible and can be adapted to different national nuclear policy and needs.

The most mature technology, the sodium cooled fast reactor, has more than 400 reactor years of experience. Experimental, prototype, demonstration and commercial units are already operating in a number of countries, including our host country Russian Federation.

Ladies and Gentlemen,

I am pleased that so many experts from all over the world have come together for this event to share their knowledge and expertise.

I have no doubt that this Conference will move us a step closer to helping innovative nuclear energy systems become viable for industrial deployment in the coming decades.

I wish you every success in your discussions.

Thank you.

2.3. HONORARY GENERAL CHAIR

Opening speech as provided, verbatim.

Fast Reactor Development and International Cooperation

Subhash Chandra Chetal

Ex-Director, Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam, India

Introduction

Let me at the outset express my deep appreciation for Russian fast reactor programme for having demonstrated excellent and consistent performance of BN-600 sodium cooled reactor giving confidence that fast reactors can provide capacity factor comparable to thermal reactors and for experimental reactor BOR-60 for providing valuable data for both Russian programme and making the reactor available for international cooperation. I wish the performance of BN-800, the highest rated sodium cooled fast reactor in operation presently would exceed performance of BN-600. The fast reactor community is looking forward for an early start of construction of a number of Russian fast reactors namely sodium cooled fast reactor BN-1200, lead cooled demonstration fast reactor BREST-300, experimental lead bismuth reactor SVBR-100 and multipurpose experimental fast reactor MBIR.

The fast reactors are the ideal complement to currently operating water reactors in order to ensure a sustainable nuclear resource able to supply energy for future generations, while optimising fuel cycle and management of the radioactive waste.

Sodium Cooled Fast Reactors

The fast reactor community is well aware that there is no perfect coolant for fast reactors and the choice gets deliberated from time to time both within the individual countries as well as in international forums. This is the reason for a number of variants of fast reactors in different countries and Gen IV programme. Sodium coolant, in spite of a few distinct disadvantages, has been the most favoured one since beginning of the fast reactor programme in different parts of the world. Considerable operating experience of over 400 reactor years is available from 20 sodium cooled fast reactors. The early reactors were low power research reactors and the highest rated experimental reactor was 400 MW(th) Fast Flux Test Facility (FFTF) which did not include electricity generation. Presently three research reactors are in operation BOR-60 in the Russia Federation, FBTR in India and CEFR in China. Demonstration reactors were built in different parts of the world; Phenix in France which operated for 36 years, BN-350 in Kazakhstan which operated for 27 years, PFR in UK which operated for 20 years. BN-600 is the only one in operation among the fast reactors that were built during 70s and is in operation since 1980. Monju reactor in Japan operated for a short period and remained shut down for over 15 years due to moderate sodium leak in the secondary circuit and restarted for a short period before permanent shutdown. Superphenix, 1200 MW, and the highest rated fast reactor built so far had to be shut down prematurely after just 12 years due to state coalition politics. BN-800 is the only power reactor presently in operation in the world.

Very valuable operating experience has been accumulated from the operating fast reactors as a feedback for future design in terms of both worth retaining and discarding for future designs in terms of materials, design options and sodium technology. The experience is getting shared as a part of IAEA meetings or through bilateral collaborations. As regards to fuel, mix oxide, metal and carbide fuel have shown excellent results. The fuel burnup has been increased systematically with operating period with improvement in core structural materials. In the case of FBTR carbide fuel reactor, burnup has been increased progressively without change of cladding materials based on post irradiation examination as the initial burn up was fixed on conservative basis in absence of fuel swelling data. Oxide dispersion strengthened (ODS) for cladding material is under development in a few countries for enhancement of burnup. It is likely that reprocessing considerations could dictate the choice of ODS in case of Purex process. Sodium leaks from use of stabilised stainless steel 321 in Phenix and PFR due to reheat cracking, and damage to reheater in stainless steel due to caustic stress corrosion cracking in PFR has led to the rejection of stabilised grades 321 and 347 for sodium systems of future reactors and elimination of austenitic stainless steel used earlier for superheater and reheater. Except for Phenix, operation of Intermediate Heat Exchangers (IHXs) has been excellent. Secondary sodium leak from outlet header in Phenix has been well understood and the design improvements are well reflected in future designs. Operation of sodium pumps has been exceptionally well in all the reactors except for early difficulties in BN-600 pump drive and oil leakage from the PFR. Operation of steam generators has shown that this component holds the key to plant capacity factor. There is nothing common in the various designs of steam generators even in the same country: for example, the design of steam generators in France ranging from Phenix modular construction to monolithic unit in Superphenix to EFR and now ASTRID or the example of steam generators from BN-800 to BN-1200. Though operating experience from EBR II of double wall steam generators has been free from any tube leaks, there is no guarantee that tube leak cannot take place leading to sodium water reaction and thus some designers have retained single wall tube design due to economics. Concerns of sodium water reaction led to the study of option of gas in the power conversion system of ASTRID in France. Core subassembly handling had impacted FBTR in the early period due to operation errors, and also severely impacted both Japanese reactors – Monju and Joyo. Lessons learnt from fuel handling incident of FBTR has led to smooth operation till date. The causes of sodium leaks in different reactors due to design inadequacy in detailing the design, manufacturing deficiency, materials of construction and thermal striping are fairly well understood and are getting incorporated in future designs. A serious concern is still felt in some countries concerning sodium leaks, and double wall piping in secondary circuits is selected to avoid sodium fire by a few designers.

The lessons learnt from the operating experience have also shown the path for an optimum number of components from both economics and capacity factors.

Presently in India, 500 MW oxide fuel reactor is under advanced stage of commissioning.

Sodium Cooled Fast Reactors currently under design

The reactors under design are BN-1200 in Russia, ASTRID in France, FBR1/2 in India, PGSFR in Korea and one in China. All these have retained traditional sodium cooled reactors except for ASTRID where the option is still open for the power conversion system.

Design of fast reactors, in particular after Fukushima accident, has led to adoption of additional features addressing the emerging safety requirements in terms of design extension, practical elimination, passive systems for decay heat removal and shutdown systems, more conservative design parameters for external events, emphasis on in-service inspection and minimising the consequences of whole core accident.

BN-1200 reactor has a few distinct design features in comparison to earlier reactors BN-600 and BN-800 in terms of fuel as nitride, design life as 60 years, bigger pin diameter, single enrichment instead of three and increased burn up, four secondary loops, integrated once through steam generator, primary sodium purification inside the pool and decay heat removal directly from the primary pool and storage of spent fuel in water. These options reduce specific steel requirements in comparison to the previous designs, and higher capacity factor of 90% thus providing much better economic parameters.

ASTRID, a 600 MW demonstration reactor, is being designed with enhanced safety features of heterogeneous core with larger pin diameter to reduce the sodium void coefficient, third level of shutdown, enhanced inspectability of permanent in-core components, considering core melt down accident and modular steam generator with design based on rupture of all the tubes in case traditional water steam cycle is chosen.

The Korean PGSFR, a Gen IV reactor of 150 MW is under design with pool configuration, metallic fuel with core outlet temperature of 545°C, two pumps and four intermediate heat exchangers with number of reactor components planned in grade 91 steel.

FBR1/2 of 600 MW each represents the first step towards a design with twin unit in India in line with the approach for PHWRs to have the economic advantage with design features in comparison to PFBR along with improved economics and enhanced safety. The design features of FBR1/2 are core design with reduced sodium void coefficient, increase in primary pumps from two to three retaining two secondary loops with two IHXs and one secondary pump each, reduced number of steam generators from four to three per loop with longer tube length, in vessel purification, and secondary shutdown system as passive one. An ultimate shutdown system is also planned based on either liquid poison system on boron carbide granules.

India is also designing an experimental metal fuel reactor which will replace FBTR. This reactor of 100 MW(th) will employ full length metal fuel subassembly similar to 1000 MW(e) power reactor. To start with, experimental metal fuel pins are under irradiation in FBTR.

Lead Bismuth/Lead Cooled Fast Reactors

Lead bismuth fast reactors were used in Russian submarines of the 1970s as these were significantly lighter than water cooled reactors due to higher efficiency. Lead bismuth or lead coolant possesses significant advantage over sodium in terms of chemical activity both with air and water thus minimising the concerns of sodium fires and violent sodium water reaction. Lead bismuth produces a considerable amount of polonium, a highly radioactive substance posing a severe maintenance issue while lead cooled system produces orders of magnitude less polonium as well as being much cheaper than lead bismuth and is hence being favoured in spite of very high melting point of 327°C against 127°C for lead bismuth. Corrosion control with lead or lead bismuth coolant is far more challenging than that of sodium coolant.

Construction of BREST, a 300 MW(e) lead cooled reactor with nitride fuel is expected to start soon in the Russia Federation.

ALFRED, a 125 MW technology demonstrator, is under R&D stage and finalization of European partners for its construction.

In Russia, design of 100 MW lead bismuth fast reactor SVBR, with flexible fuel design, primarily based on Russian submarine experience, is being pursued.

Gas Cooled Fast Reactors

Gas cooled fast reactor overcomes the distinct disadvantages with sodium of chemical incompatibility with air and water, limitation in core outlet temperature to avoid sodium boiling and opacity. However, gas cooled reactor because of poor thermal inertia will demand very reliable decay heat removal system and shutdown systems. Significant development in fuel element design will be needed to exploit high core outlet temperature of 850°C.

In the near future, only an experimental reactor of Gen IV design of 75-100 MW thermal rating is planned in Europe, initially at low core outlet temperature with MOX fuel and then with ceramic fuel for 850°C core outlet temperature.

Small and Very Small Reactors

Small and very small size reactors could be utilized in remote locations where localised power is required for energy intensive industrial uses like mining, military or municipal applications. These reactors of modular construction factory built could lower capital cost and provide quicker returns on investment. Also, the simpler design with inherent safety features would enhance more potential vendors. A few examples in this category are Super Safe, Small & Simple 4S nuclear battery a joint venture of Japanese and US industries with features of metal fuel sodium cooled reactor, factory built, capable of three decades of continuous operation without on-site refuelling designed in two different power ratings of 10 and 50 MW. Another variant in this category is Small Sealed Transportable Autonomous Reactor SSTAR, a lead bismuth or lead as coolant of 20 MW rating. SEALER, Swedish advanced lead cooled reactor designed for use by mining industry in Canada is the lowest rated fast reactor of just 3 MW and is under licensing process in Canada.

USA which had built a number of research reactors in the earlier period has no reactor construction planned in the near horizon and is focussed on systems, materials and safety analysis.

Economics

Since only sodium cooled fast reactors had been built for demonstration and commercial purposes, one can say about economics of SFRs only. The technology viability of SFRs has been well demonstrated in experimental and demonstration reactors. However, the economic competitiveness of SFR has not been well proved yet. The perceived higher cost of SFRs compared with LWRs has impeded its growth. The economic comparison of SFR vs LWR has strong linkages with the cost of uranium which presently puts pressure on SFRs to look for ways and means to improve on capital cost, construction time and capacity factor.

It is a matter of great satisfaction that cost of BN-600 has been found to be comparable to VVER after considering economy of scale. For BN-1200 reactor also, the economic comparison with VVER is attractive.

Studies for European Fast Reactor (EFR) showed that there existed considerable scope for capital cost reduction over Superphenix design.

SFRs, unlike LWRs, have not yet reached the stage of economic advantage of multiple reactors at site and learning curve as every reactor design even in the same country is far different from the previously built reactors.

There is a strong need to arrive at the most economical design based on life cycle cost as regards to optimum number of components and systems.

It is time for countries building the next fast reactors to repeat as much as possible of the previously built to reduce engineering and R&D costs and construction time.

Let me again emphasize that large scale deployment of SFRs will be feasible only when economic parameters are attractive. The same will hold good for other types of fast reactors.

The countries pursuing a close fuel cycle need to give serious consideration to cost of reprocessing plants. Indian assessment shows that it is prudent to co-locate the fast reactor fuel cycle facility for reprocessing and refabrication at the same site as the reactor and should be designed for multiple reactors. In light of this, Fast reactor Fuel Cycle Facility is under construction at the site of PFBR to reprocess and refabricate the MOX fuel elements for PFBR and two more MOX reactors of 600 MW each.

Licensing

First-of-a-kind reactors in the earlier period were far easy to be licensed than the present times. Early designs were reviewed by experts and not on very rigid regulations. A fair amount of testing was done in support of the licensing process. Power increase was gradual, and designers did their best to accommodate safety issues raised in previous reactor licensing issues. One of the challenges in licensing of fast reactors is the background of experts in the regulatory bodies. Thermal reactor regulations are matured, and experts predominantly belong to thermal reactors. In spite of significant differences between fast reactors and thermal reactors, non-conclusive discussions do take place on issues like loss of coolant accident, containment related issues and comparing the design safety features against the thermal reactor safety requirements, risking delay in licensing process. These issues are expected to be of relatively less concerns in case of the next reactors, but challenges continue for first-of-a-kind reactors in every case.

International Cooperation

International cooperation is of extreme importance at the present juncture for fast reactors for efficient promotion and maximisation of the results using resources of R&D institutes, contribution to ensuring nuclear safety and support for developing human resources in the nuclear field to new entrants to the fast reactors. The international cooperation facilitates developing international standards of nuclear safety which are asset to designers, regulators and the public at large. Let me now touch upon a few examples of international cooperation in the field of fast reactors.

The Generation IV International Forum (GIF) is a cooperative international endeavour which was set up in 2001 to carry out the R&D needed to establish the feasibility and performance capabilities of the next generation nuclear energy systems. The GIF has 14 members. The goals adopted by GIF provided the basis for identifying and selecting six nuclear energy systems for further deployment. The systems selected include both thermal and fast reactors. Depending on their respective degree of technical maturity, the first Gen IV systems are expected to be deployed commercially beginning in year 2030. The six reactor systems identified are:

- Sodium cooled reactor;
- Lead cooled reactor;
- Gas cooled reactor;
- Molten salt reactor;
- High or very high temperature reactor;
- Supercritical water cooled reactor.

One of the important document of vital importance to the sodium cooled fast reactor community issued by GIF task force is the safety design criteria for Gen IV reactor system in the year 2012.

Another programme with similar aims as GIF is the INPRO Innovative Nuclear Reactors and Fuel Cycle programme coordinated by IAEA. INPRO was established in 2000 to help ensure that nuclear energy is available to contribute to meeting the energy needs of the 21st century in a sustainable manner. INPRO forum brings together technology holders and users so that they can consider jointly the international and national actions required for achieving desired innovations in nuclear reactors and fuel cycle. INPRO membership currently consist of 41 members. A number of INPRO collaborative projects have been completed. These include the case study of BN-800 reactor and decay heat removal for liquid metal fast reactors.

Let me now touch upon the European Sustainable Nuclear Industrial Initiative (ESNII). This initiative primarily addresses the need for demonstration of Gen IV reactor technologies together with supporting research infrastructure, R&D work and fuel facilities. The specific reactors under the current activities are:

ASTRID sodium cooled reactor as the reference solution with France as the leading nation.

As an alternative technology, the lead cooled fast reactor ALFRED with the construction of an experimental reactor in Europe to demonstrate the technology and supported by lead bismuth irradiation project MYRRHA in Belgium. With MYRRHA, Europe will again operate a fast neutron irradiation facility in support of technology development in particular for materials and fuel irradiation tests of three fast reactors SFR, LFR and GFR.

As a second alternative technology, the gas cooled fast reactor ALLEGRO experimental reactor.

Bilateral agreement of Terra power, a US company with China National Nuclear Cooperation signed in 2015 to develop an innovative travelling and now stationary wave reactor based on design of metal fuel sodium cooled reactor with initial plan to build a demonstration plant of 600 MW followed by large commercial plant of 1150 MW is being watched by the fast reactor community with interest for its implementation.

Every country pursuing fast reactor programme has a number of bilateral R&D cooperation agreements with other partners which is of mutual benefit to both countries.

International cooperation in fast reactor development in terms of International Conferences on fast reactors and associated fuel cycles like the present one being organised from today in this lovely place, annual meetings of TWGFR, technical theme meetings and a number of TWG-FR publications are other forms of international cooperation thanks to IAEA for the benefit of both specialised and new entrants to the fast reactors.

The new entrants to fast reactors like the case of Republic of Korea for their first experimental sodium cooled fast reactor PGSFR would be wishing for international support for irradiation data of cladding materials, experimental testing of important sodium components as well as manufacturing of components for the reactor. It is important that on mutually agreed terms, such technical support on commercial terms could be implemented as every reactor provides valuable data to entire fast reactor community.

Closure

Let me close my talk wishing large scale deployment of sodium cooled fast reactors, the only one available presently for power generation, success in other technologies of lead and gas cooled fast reactors for deployment in the long term and enhanced indispensable international cooperation.

3. SUMMARY OF THE PLENARY SESSIONS

This section provides the summary of plenary presentations made by China, France, India, Japan, Republic of Korea, the Russian Federation, European Commission, Generation IV International Forum and International Atomic Energy Agency. The Russian Federation as a host country delivered an additional first keynote speech.

Full unedited keynote contributions are provided after the short summary of the plenary sessions.

3.1. RUSSIAN FEDERATION

E. Adamov, Former Head of the **Russian Atomic Energy Ministry (ROSATOM)**, presented the results obtained during the five years of the PRORYV Project, which confirmed the technological feasibility of its fundamental principles and made it possible to proceed to the practical development stage and to a new nuclear technological platform. The implementation of the developed design, engineering and processing solutions and the realization of the planned Pilot Demonstration Energy Complex R&D programme (phase one start-up scheduled for 2020) made it possible to anticipate, with confidence, the development of a prototypical industrial energy complex capable of operating within a two-component nuclear power scheme by 2030. In this regard, a prototype of a next generation nuclear reactor is to be built at the site of Joint Stock Company Siberian Chemical Plant by the joint efforts of ROSATOM experts and Russian Academy of Science universities and institutes.

3.2. CHINA

H. Yu, Deputy Director responsible for International Cooperation, **China Institute of Atomic Energy (CIAE)**, explained that China is developing nuclear energy with a capacity of about 30 GW, with 35 NPPs in operation and another 19 under construction, with the target to reach 58 GW by 2020 and about 400-500 GW by 2050, although the average utilization of nuclear power plants in China has declined for the last three years. The reasons behind this decline are that China's economy is in a period of adjustment, with cheaper coal prices and an abundance of water and hydroelectric power. Several new generation nuclear energy systems are currently under study in China with the focus on sustainable nuclear fuel cycles, that save uranium resource and reduce the burden of generated radioactive waste, to meet the future demands.

3.3. FRANCE

S. Pivet, Deputy Head of the Nuclear Energy Division at the **French Alternative Energies and Atomic Energy Commission (CEA)**, explained that nuclear energy will remain one of the pillars of the future French low carbon energy mix. The closed fuel cycle associated with fast neutron reactors will lead to drastic improvement in uranium resources management and important reduction in footprint and radiotoxicity of the final wastes. The French programme on Gen IV is based on the “Accompagnement Spécifique des Travaux de Recherches et d’Innovation Défense” (specific support for defence research projects and innovation) programme, with the basic design phase ongoing for the period 2016–2019. The schedule and organization of the next phases are under preparation by the French government and industrial partners. At the same time, France is conducting an active survey on other Gen IV fast and thermal neutrons systems.

3.4. INDIA

A. Bhaduri, Director of the **Indira Gandhi Centre for Atomic Research (IGCAR)**, summarized the details of the Indian fast reactor programme and discussed its status and R&D achievements. The fast reactor programme in India has several aspects. The construction of the Fast Breeder Test Reactor (FBTR) allowed comprehensive experience in construction and operation, and in the provision of material irradiation data, including for reactor and energy conversion systems. The Prototype Fast Breeder Reactor (PFBR) is intended for technical and economic demonstration of the system. The fast reactor programme will provide additional R&D opportunities in design and fuel reprocessing, as well as in improved economics and enhanced safety. The Indian fast reactor programme is essential for the security and sustainability of energy in India. Experience from FBTR operation and PFBR design, manufacture, construction and safety review have improved confidence in fast reactor deployment in a closed fuel cycle mode and with no technological constraints envisaged. In striving for a higher growth rate, R&D on metal fuel with high breeding potential along with associated fuel cycle technologies is in progress. India continues to place a strong emphasis on R&D directed towards building up a substantial fast reactor programme in the future.

3.5. JAPAN

Y. Sagayama, **Japan Atomic Energy Agency (JAEA)** announced that “The Fourth Strategic Energy Plan” of Japan was approved by the Cabinet in April 2014: it states that nuclear energy is an important baseload power source as a low carbon and quasi-domestic source even after the TEPCO Fukushima Dai-ichi accident. Japan will promote nuclear closed fuel cycle in terms of the efficient use of resources and volume reduction. Mitigation of degree of harmfulness of high-level radioactive waste carries out fast reactor cycle R&D for the commercialization, taking advantage of international cooperation.

3.6. REPUBLIC OF KOREA

J. Yoo, Technical Director of the Sodium Cooled Fast Reactor Development Agency(SFRA), Korea Atomic Energy Research Institute(KAERI) informed that the Korea Atomic Energy Commission authorized the R&D action plan for the development of the advanced sodium cooled fast reactor (SFR) and the pyro-processing technologies to provide a consistent direction to long term R&D activities in December 2008. This long term advanced SFR R&D plan was revised by Korea Atomic Energy Promotion Council (KAEPC) in November 2011 in order to refine the plan and to consider the available budget for SFRs. The revised milestones include the specific design of a prototype SFR by 2017, specific design approval by 2020, and construction of a prototype Gen IV SFR (PGSFR) by 2028. The prototype SFR programme includes the overall system engineering for SFR system design and optimization, integral V&V tests, and major components development. Based upon the experiences gained during the development of the conceptual designs for Korea Advanced Liquid Metal Reactor (KALIMER), the conceptual design of PGSFR had been carried out in 2012 following by performing a preliminary design since 2013. The first phase of the development of PGSFR had been completed at the end of February 2016 and the second design phase at the end of 2016. All the design concepts of systems, structures and components (SSCs) were determined and incorporated into the Preliminary Safety Information Document (PSID), which includes basic design requirements, system and component descriptions, and the results of safety analysis for the representative accident scenarios. The PSID will be a base material for a pre-review of the PGSFR safety. The target of the second phase of PGSFR design was to prepare a Specific Design Safety Analysis Report (SDSAR) by the end of 2017. The SDSAR is equivalent to the conventional Preliminary Safety Analysis Report (PSAR) but without the specific site information of the plant. To support the design, various R&D activities are being performed in parallel with design activities, including V&Vs of design codes and system performance tests.

3.7. RUSSIAN FEDERATION

A. Tuzov, Director of the **Research Institute of Atomic Reactors (RIAR)**, delivered a presentation on the Russian Federation's research and pilot fast reactors, which are considered as the basis for the development of the commercial reactor technologies. Research reactors have demonstrated their role as the crucial element for the consistent and coherent development of fast reactor technology. Sodium fast reactor technology is currently the only mature concept implemented on an industrial scale. The lessons learned confirm the appropriateness of the step by step approach, on the road from reactor test facilities to prototypes and, ultimately, commercially feasible fast reactors. The viability of the whole fast reactor technology clearly depends on ensuring the closure of the fuel cycle. Commercially viable long term solutions will require full usage of existing research reactor infrastructure. The ageing of the sophisticated infrastructure should demand consolidation of stakeholder efforts, also at the international level. Considering the RIAR research infrastructure and the personnel qualifications at the institute, as well as the many years of experience gained in the development and operation of fast reactors and related fuel cycle facilities, RIAR fulfills its role admirably.

3.8. EUROPEAN COMMISSION (EC)

J-P. Glatz, Head of **Nuclear Safety of the JRC Directorate for Nuclear Safety and Security of the European Commission (EC)**, discussed contributions for the development of fast reactor systems based on platforms, initiatives and alliances created to facilitate distribution of the R&D resources. The activities are organized in multi annual framework programmes (at present Horizon 2020 Project) with collaboration projects co-financed by Directorate General for Research and Innovation (DG RTD) and with direct research carried out in the Joint Research Center (JRC), new EC Directorate for Nuclear Safety and Security. These programmes focus mainly on the reactor itself (reference: SFR, alternatives: LFR and GFR), as well as fuels, fuel cycles and related materials. Furthermore, JRC is the implementing agent for European Atomic Energy Community (EURATOM) in GIF and in this capacity contributes to all six GIF reactor conceptual designs selected for further R&D, taskforces and cross-cutting working groups.

3.9. GENERATION IV INTERNATIONAL FORUM (GIF)

F. Gauché, Head of **Generation IV International Forum (GIF)**, introduced and gave details of the recent GIF activities over the past four years. New members have joined GIF since FR13 conference in Paris. Detailed descriptions of six GEN IV conceptual reactor designs were presented, compared and discussed. SFRs remain in focus to GIF members and also attract the interest of the private sector. Recently, GIF achieved significant success in developing Safety Design Criteria (SDC) and Safety Design Guidelines (SDG) for SFRs. The recent GIF reports on SDC and SDG became a topic for the panel discussion at the FR17 conference. Other reactor designs, such as LFR, MSR and GFR are also studied under the GIF framework. GIF established an education and training task force to promote GEN IV systems and related topics.

3.10. OECD/NUCLEAR ENERGY AGENCY (NEA)

T. Ivanova, Head of Division of Nuclear Science, **Nuclear Energy Agency of Organization for Economic Cooperation and Development (OECD/NEA)**, highlighted the commitment of the NEA to support advances in the fundamental science and technology that underpin fast reactors, including nuclear data, integral experiments, transient benchmarking, materials science and fuel cycle scenarios and technology, serving as a forum for exchanging information and promoting collaborative activities. As an example, the group on the safety of advanced reactors (GSAR) has been recently established to discuss regulatory and safety issues related to Gen IV designs and, during the workshop on reactor systems and future energy market needs, the future of nuclear baseload was extensively debated.

3.11. INTERNATIONAL ATOMIC ENERGY AGENCY (IAEA)/ INTERNATIONAL PROJECT ON INNOVATIVE NUCLEAR REACTORS AND FUEL CYCLES (INPRO)

J. Philips, Section Head of the IAEA **International Project on Innovative Nuclear Reactors and Fuel Cycles** (**INPRO)**, covered the main activities of INPRO in the areas of nuclear energy system sustainability assessment and in whole system scenario analysis in support of long term planning for sustainable development of nuclear energy. The special focus of this presentation was on the projects that directly involve fast reactors, related fuel cycles and the potential for international cooperation through trade and R&D collaboration. In particular, projects include the application of the INPRO methodology for assessment of the sustainability of specific sodium fast reactor designs in China, India and the Russian Federation in areas of safety and economics. In addition, the whole system scenario analysis of the regional and global nuclear energy systems involving fast reactors, related fuel cycles and trade-in related nuclear products and services between cooperating countries were also discussed. Services provided to the Member States in the use of the INPRO methodology and scenario analysis tools were also described.

4. PLENARY SESSION - KEYNOTE PAPERS

4.1. RUSSIAN FEDERATION

Closed Fuel Cycle Technologies Based on Fast Reactors as the Corner Stone for Sustainable Development of Nuclear Power

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Abstract. This article analyses problems and approaches to modern nuclear power development using closed nuclear fuel cycle and fast reactors. It describes specified technical requirements for nuclear power systems in large-scale nuclear power industry. Targets and scientific problems solved by ROSATOM’s “PRORYV” Project which is a part of the Federal State Programme “Nuclear Power Technologies of New Generation in the Period of 2010-2015 and up to 2020” are examined.

Key words: nuclear reactor, closed nuclear fuel cycle, nitride fuel, radioactive waste.

1. INTRODUCTION

The concept of nuclear power development in the Russian Federation that was clearly stated in the year 2000 document [1] and further developed in papers [2, 3] suggests the development and introduction of fast reactors with inherent safety and closed nuclear fuel cycle as a priority target. After the period of conceptual R&D at the turn of 2012, the PRORYV Project [4, 5] started first practical steps to the implementation of the concept and creation of the new technological base for large-scale use of nuclear power technologies.

Analysis of the results achieved within the frameworks of this project during the past 4 years since FR13 conference [5] are presented in this paper.

2. GOALS AND TASKS OF THE PROJECT

PRORYV Project suggests the development of fast reactors with inherent safety and on-site closed nuclear fuel cycle that should meet the following requirements:

- Elimination of accidents requiring population evacuation and resettling;
- Maximum possible use of energy potential of uranium resources;
- A gradual approach to radiation equivalent (compared to natural raw materials) of Radioactive Waste (RAW) disposal;
- Technological support to non-proliferation;
- Competitiveness of nuclear power compared to other means of energy generation.

Experimental testing and demonstration of new technological solutions are planned to be carried out at the Pilot Demonstrator Energy Complex (PDEC) in Tomsk. It consists of pilot demonstrational power unit with BREST-OD-300 lead cooled reactor operating with mixed nitride fuel, fuel refabrication and fuel recycling facilities.

The project of power unit with BN-1200 reactor should prove that new reactors can be competitive to the best NPPs on thermal neutrons. The project uses as much as possible all knowledge gained during the development and operation of BN-600 and BN-800 reactors. At the same time, a commercial project of BN-1200 should be highly innovative and apply new technical solutions that will ensure fulfilment of above mentioned requirements of inherent safety, excluding those that are naturally tied with the use of sodium coolant.

3. PILOT DEMONSTRATOR ENERGY COMPLEX PROJECT

Pilot Demonstrator Energy Complex (PDEC) project should be the first in the world to demonstrate technical-economical parameters of the whole set of facilities of the Closed Nuclear Fuel Cycle (CNFC) at one site.

The on-site CNFC allows perfecting "short fuel cycle" technology with a minimum Spent Fuel (SF) cooling before reprocessing. Table 1 shows the main PDEC parameters, and Fig. 1 a general scheme of phased commissioning.

It should be noted that all the developed technologies (excluding logistics) and components can be used also for centralized arrangement of a fuel cycle provided the optimum technical and economic efficiency is attained with acceptable logistics. Module organization of production facilities seems to be the most logical one.

TABLE 1. MAIN PDEC PARAMETERS

Rated output power, gross	300 MW
Fuel type	Mixed U-Pu nitride (MNIT)
Design lifetime, year	30
Design lifetime of fuel cycle equipment, year	30
Fuel fabrication/refabrication capacity, t/year	14.75
Spent fuel reprocessing capacity, t/year	5

3.1 BREST-OD-300 Reactor design and power unit on its basis

In 2016, the basic design of the innovative BREST-OD-300 lead cooled reactor facility with mixed nitride fuel was finalized. Its main characteristics are shown in Table 2 and the overall outlook in Fig. 2. Main parameters of the power unit (its design is also completed) are shown in Table 3 and its view is presented in Fig. 3.

TABLE 2. BREST-OD-300 MAIN PARAMETERS

Thermal power, MW	700
Number of FA in the core	169
Fuel	MNIT
Fuel load, t	20.6
Fuel load, t	20.6
Breeding Ratio (BR)	1.05
Number of loops	4
Primary coolant	Lead
Maximum coolant pressure in the primary circuit, MPa	1.17
Coolant temperature at the core inlet/outlet, °C	420 / 535
Average temperature of SG working medium, °C	340 / 505
SG outlet pressure, MPa	17
Steam output, t/h	1500

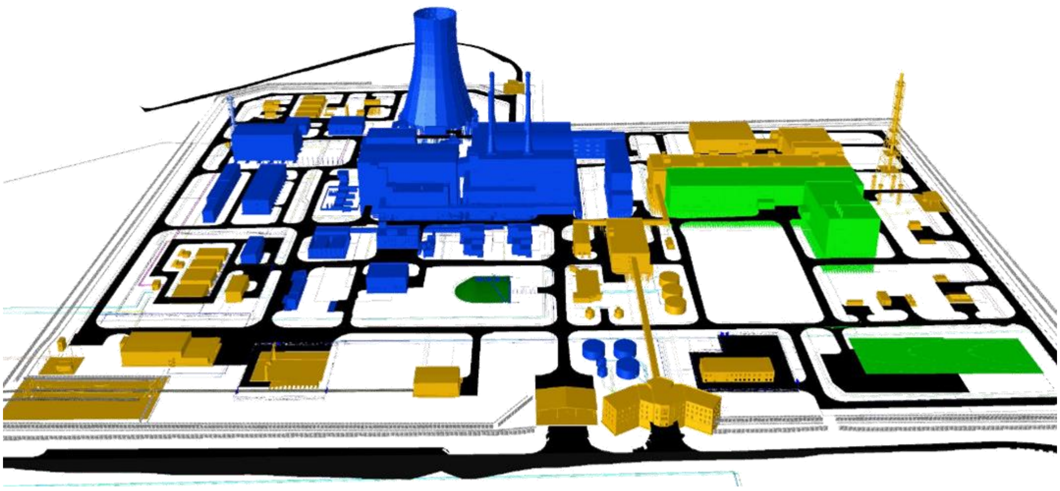


FIG. 1. Pilot Demonstrator Energy Complex (PDEC) general scheme of phased commissioning: yellow – fabrication/refabrication facility – first stage, blue – BREST-OD-300 reactor unit – second stage, green – SF recycling and RAW handling facility.

The use of non-boiling and not interacting with water and air lead coolant allowed implementing double circuit reactor layout that significantly differs from the traditional three-circuit design of sodium cooled reactors and therefore improves technical-economical parameters of the liquid metal cooled reactor. Special reports at the Conference are dedicated to the state of development of this power unit and related equipment.

Power unit design has passed the State Expert Assessment and is filed to Rostechnadzor to be licensed for construction.

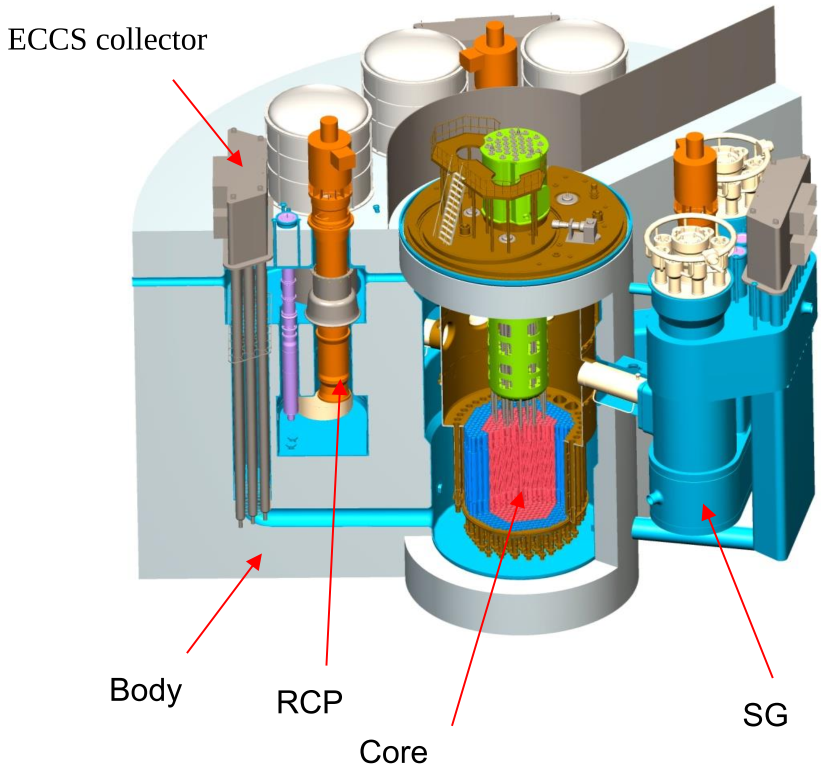


FIG. 2. Integral design scheme of BREST-OD-300 reactor unit.

TABLE 3. MAIN PARAMETERS OF BREST-OD-300 POWER UNIT

Parameter	Value
Reactor thermal power, MW	700
Turbogenerator electrical power, MW	300
Efficiency (gross), %	43
Capacity factor	0.8
Reactor service life, year	30
Design-basis seismic, point	7

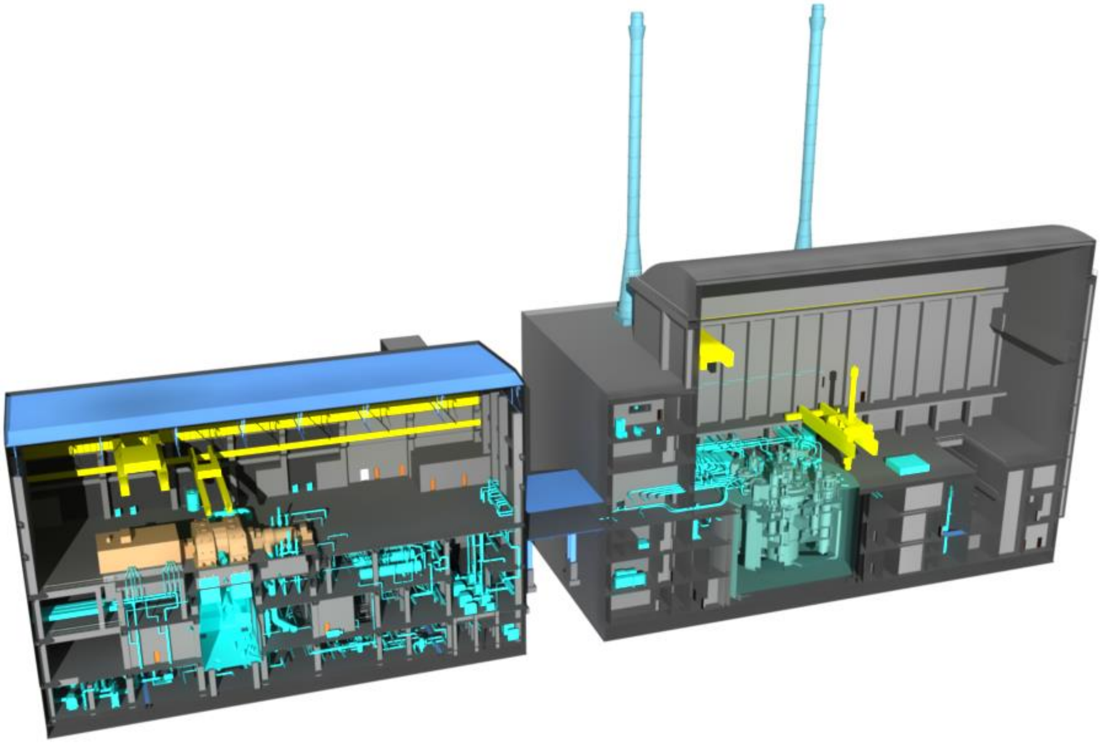


FIG. 3. Power unit view (left – central hall, right – turbine hall).

3.2. On-site fuel cycle design

On-site fuel cycle consists of two main facilities – Fuel Fabrication Facility (FFF) and Fuel Recycling Facility (FRF), that includes radioactive waste handling system. At the first facility, a pilot demonstration production of mixed nitride fuel with plutonium and depleted uranium was created for the first time in the world with the use of carbothermal preparation technology.

The single facility can work with both raw materials and BREST-OD-300 spent fuel (SF) recycling products. It also suggests including Minor Actinides (MA) into fuel for further transmutation. FFF is under construction (Fig. 4) as the first PDEC construction stage with commissioning in 2020.

It's worth noting that an alternative technology of direct hydration is at R&D stage within the frameworks of PRORYV Project as well. Gradual implementation of this technology is considered: the first stage – hydrometallurgical treatment with further introduction of the combined option that includes pyro-chemical processing at the initial stage and further purification by hydro-metallurgical techniques. The option of switching to single pyro-chemical technology is also considered in case R&D results will prove the required purification goals can be met.

FIG. 4. General view of PDEC construction site.

3.3 Major Tasks for R&D at PDEC

Construction of PDEC will allow demonstrating the capacity of technologies based on fast reactors (FR) with inherent safety in CNFC and conducting a series of R&D cycle that will open the way to their implementation in an industrial scale, including:

- Demonstration of BREST-OD-300 operation with the minimum reactivity margin that eliminates the possibility of reactivity-induced accidents and optimization of BREST-OD-300 reaching of the equilibrium mode;
- Verification and demonstration of full fuel Core Breeding Ratio (CBR~1) and utilization of full power potential of natural uranium without the use of blankets;
- Mastering of the lead coolant technology in the real conditions of large NPP;
- Testing of innovative technical solutions for reactor unit, FFF, FRF and acquiring of component endurance characteristics;
- Validation and demonstration of CNFC with MNIT; and
- Proof of effectiveness of MA transmutation in FR and determination of RAW characteristics in order to ensure uranium raw material radiation equivalent disposal.

4. R&D MAIN RESULTS AND INHERENT SAFETY IMPLEMENTATION IN THE PROJECT

The results of accomplished R&D showed a qualitatively new level of safety of being developed technologies (in the first turn, the reactor ones) that was specified at the initial stage by the conceptual requirements of inherent safety. The key requirement is the deterministic elimination of the possibility of severe accidents that lead to population evacuation and furthermore resettling.

4.1. Integral design of the primary circuit and elimination of accidents with loss of heat removal from the reactor core

Unlike water or gas cooled reactors, the fast reactors with liquid metal coolant are capable of deterministically eliminating the possibility of accidents with loss of coolant and/or heat removal from the core. The key part in this is played by non-boiling coolant, integral reactor design and passive systems for decay heat removal or Emergency Core Cooling Systems (ECCS) directly from the primary circuit and passive coolant flow feedback.

In the integral BREST-OD-300 design the primary coolant inventory is encased inside a multilayer metal concrete vessel (see Fig. 2). The design-basis probability of its depressurization is estimated at 10^{-9} year^{-1}.

Coolant circulation path in BREST-OD-300 is ensured by the difference in free levels. Such a scheme eliminates the need to use isolation valves. This prevents the possibility of coolant flow stop with RCPs in operation and assures continuous circulation under power loss until cooling is ensured by natural circulation through ECCS. Reactor power loss with coolant flow decrease and malfunction of active reactivity controls Unprotected Loss of Flow (ULOF) is ensured by a System of Passive Flow Feedback (SPFF) that is triggered by coolant flow rate.

All this eliminates melting of fuel cladding (the temperature reaches ~890°C just for a short time) and fuel and preserves the integrity of circulation circuit (Fig. 5). The design-basis probability of such an accident is equal to ~3×10^{-9} year^{-1}.

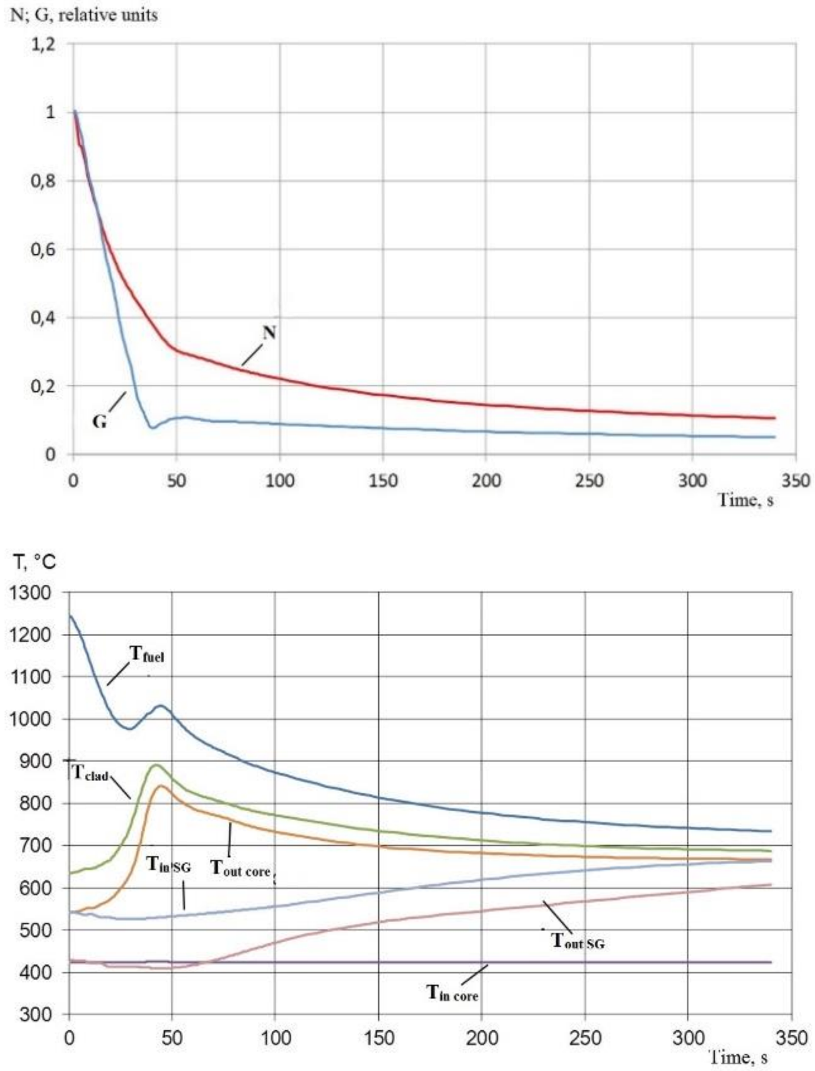


FIG. 5. History of relative reactor power and temperatures during ULOF.

4.2. Low reactivity margin and elimination of reactivity-induced accidents

FRs are capable to operate without significant reactivity change and this is their obvious advantage. Besides the absence of such effects as iodine pit, the reactivity stability during operation (fuel burnup) of an equilibrium core with CBR~1 allows eliminating the root of potential danger of uncontrolled prompt neutron runaway – the respective reactivity margin.

Design research on this idea showed that it's practically possible for implementation. In BREST-OD-300 the possibility to contain the reactivity margin of ~0.65β_{eff} (~0.23%Δk/k) during power operation is shown when using start-up load made of plutonium from PWR SF after long term cooling and reprocessing. This is basically 100 times less than thermal reactor reactivity margin and 10 times less than sodium FR of first generations (BN-350, BN-600, BN-800) margin.

Uncontrolled power ramp at the insertion of the full design reactivity margin is blocked at 1.4 N_{nom} level. At that, the temperature of fuel cladding doesn't exceed ~815°C (Fig. 6), fuel rod meltdown is ruled out. Fission product release for the first 24 hours will not exceed 6.1×10^{10} Bq (below the controlled release level during normal operation). The estimated frequency probability of such a scenario is about 3×10^{-9} year^{-1}.

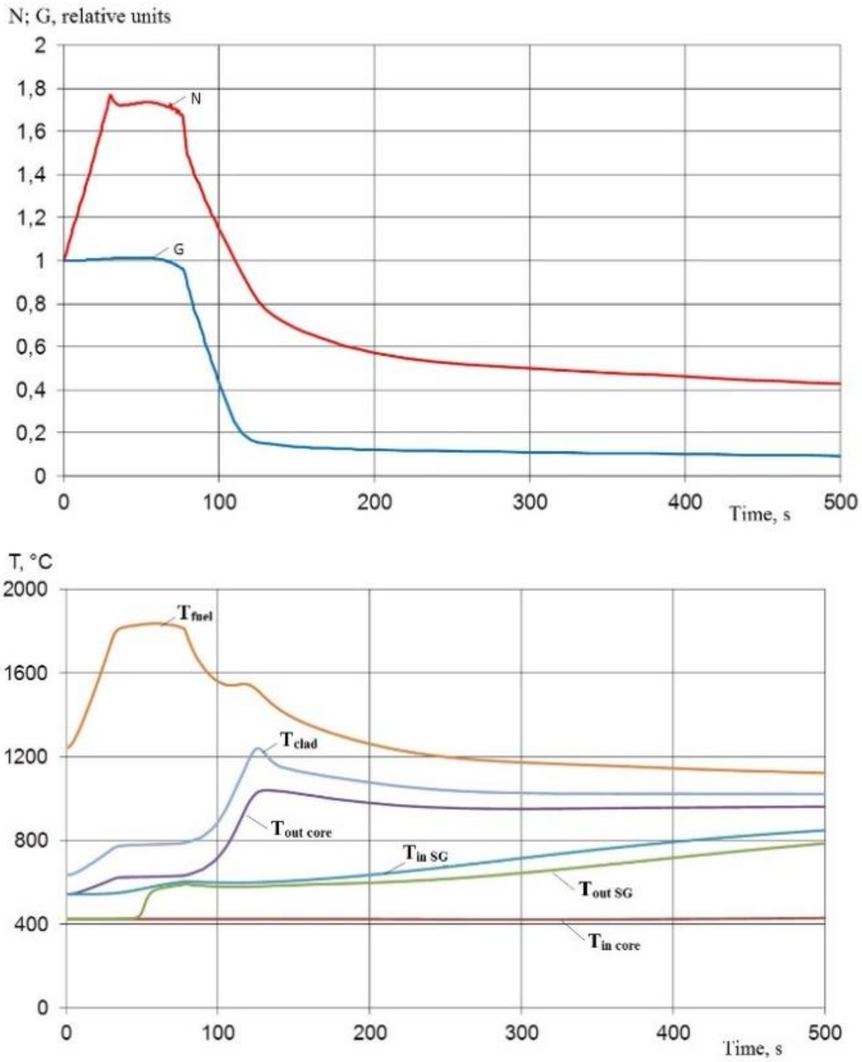


FIG. 6. Relative flowrate and reactor power and temperature history during hypothetical ULOF transient in BREST-OD-300

4.3. Approaching radiation equivalent Radioactive Waste (RAW) management

Radiation equivalent principle with its main propositions developed in the Russian Federation at the turn of the 20th century [6-9] was accepted as the ground requirement for PRORYV Project. It is shown that at certain conditions a situation is possible in nuclear power when RAW sent for final disposal in geological formations would have Potential Biological Toxicity (PBT) equal (i.e. equivalent) or less than the consumed natural uranium which means that the concept of Radiation Equivalence (RE) will be fulfilled. Radiation equivalence can be achieved at the moment of disposal or over a certain, historically short, easily forecasted period of time (200-500 years).

Radiation equivalence can be reached if transmutation fuel cycle is implemented in nuclear power with the following major constituents:

- Reprocessing of the whole volume of irradiated fuel from thermal reactors with preset fractioning for transfer of plutonium, MA and long-lived fission products to a fast reactor fuel cycle;
- Fast reactors operating in CNFC where most of actinides burnup and long-lived fission products transmutation occur while generating electricity;
- Deep purification of long-lived radioactive waste from plutonium, americium and certain other long-lived nuclides to be disposed (loss of actinides in RAW don’t exceed 0.1-0.01%); and
- Temporary storage of high active waste before final disposal for a period of 200 years in order to decrease their biological danger by one hundred times.

It is preferable to implement a new technology for uranium ore extraction that would not pollute the environment with the concurrent extraction of radium and thorium with uranium for further transmutation in FR fuel.

In order to decrease RAW long-lived radioactivity, it’s most important to remove actinides (from uranium to curium) from the material that is to be disposed. It decreases PBT of remaining fission products by 1000-10000 times. So the goal of actinides transmutation is to transfer them into fission products, but not to change one actinide into another. The contribution of main fuel nuclides in PBT of BREST irradiated fuel is shown in Fig. 7, the main one being from plutonium and americium.

^{90}Sr and ^{137}Cs with daughter nuclides are worth noting out of fission products with half-life more than 25 years. Due to small cross sections of neutron interaction, these nuclides cannot be effectively subjected to transmutation and the only way of their management is controlled storage, possibly, useful utilization of isotope devices, or disposal.

The recommended share of actinides lost in RAW of 0.1% will remain acceptable to implement the radiation equivalence until the end of this century.

To guarantee radiation equivalence of nuclear power for distant times it is necessary:

- To implement simultaneous co-extraction of radium and thorium along with uranium from the ore in the next few decades;
- To keep decreasing in the 22nd century the share of actinides lost in RAW;
- Decrease ^{14}C accumulation in fuel (switch to nitrogen enriched with ^{15}N isotope); and
- Solve the problem of transmutation of long-lived fission products.

The possibility of achieving radiation equivalence in Russian nuclear power by the end of the 21st century was shown for scenarios of nuclear power development with existing and planned thermal reactors and with a developing system of fast reactors. An example of growing power scenario is shown in Fig. 8. RAW radiation balance accumulated by the end of the 21st century in case of implemented fuel cycle with transmutation and consumed uranium raw material (uranium and its decay products, ^{226}Ra and ^{230}Th) is shown in Fig. 9. Radiation equivalence will be reached after 200 years of RAW cooling.

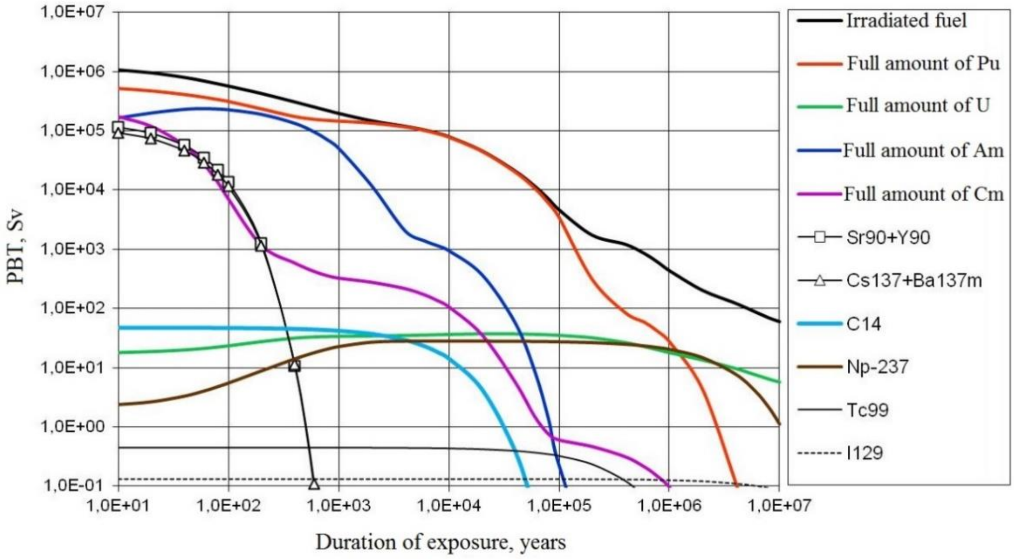


FIG. 7. Contribution of certain elements and nuclides in PBT of BREST spent fuel normalized per 1 kg of irradiated actinides (1.06 kg of nitride fuel).

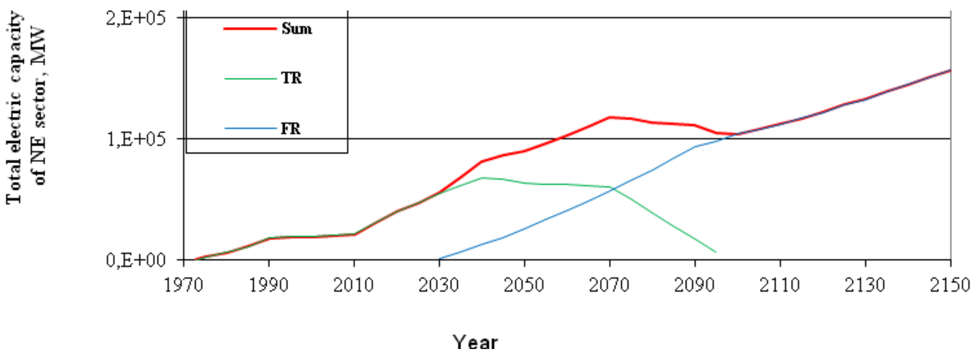


FIG. 8. Model scenario of nuclear power development in RF.

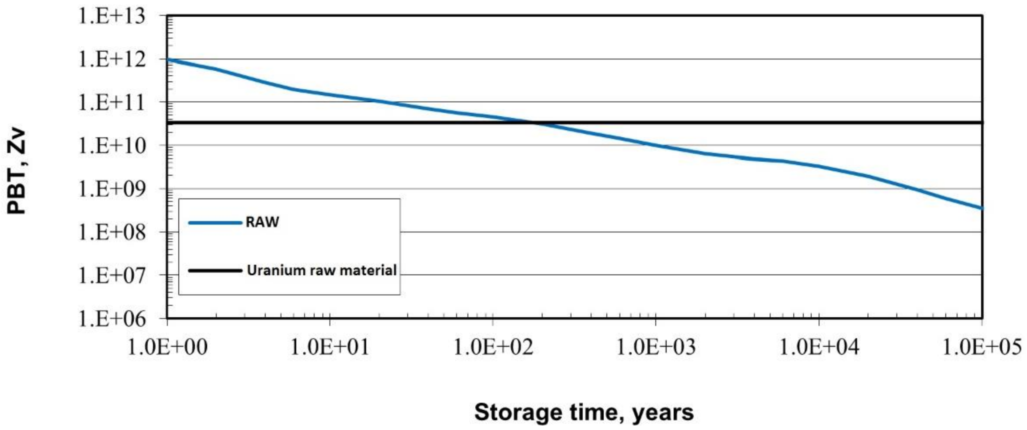


FIG. 9. Full potential biological danger of consumed raw uranium and RAW.

5. DENSE NITRIDE FUEL

5.1. Advantages of nitride fuel

At initial stages, when physical grounds and first designs of fast reactors were developed the attention was paid to reaching the highest fuel breeding ratio. This was guided by small fuel source for FRs as plutonium does not occur naturally on Earth and need to be produced with large rate. Metal fuel fits this strategy's implementation in the best way.

At present, the list of priorities defining the place of FRs in nuclear power has significantly changed. The first place is now occupied by safety problems including ecology, competitiveness, accumulation of SF and RAW, non-proliferation, optimal utilization of natural resources. Results of comparative analysis of two main types of dense fuel – nitride and metal in order to determine the best option for this strategy showed the following:

- Nitride fuel has high density and heat conductivity (1.4 and ten times higher than oxide);
- Despite metal fuel has higher theoretical density the nitride fuel is comparable to alloyed metal fuel with zirconium and higher porosity that are required to decrease swelling and increase creep resistance, which decreases its density;
- Phase changes of metal fuel and especially its interaction with steel cladding (with generation of easily melted eutectic) define low destruction margins in accidents with increasing temperature or otherwise decrease of coolant temperature is required.

— The relative disadvantage of nitride fuel is its neutron absorption in $^{14}N(n,p)^{14}C$ reaction that results in somewhat worsening of neutron balance and generation of ^{14}C with a long half-life.

The results of the research lead to a conclusion that nitride fuel is the best option. It allows reaching basically new qualities of reactor core with CBR~1 and decreasing a reactivity margin to minimum values and at the same time keep other effects and reactivity coefficients in admissible limits.

One point in favour of the metal fuel could be higher achievable fuel breeding parameters if necessary. Considered scenarios of nuclear power development in the Russian Federation and the world do not see such necessity. The fundamental point of PRORYV Project is the priority of safety and therefore the choice of CBR~1 that does not prohibit the use of blanket if higher breeding ratio is necessary (proliferation resistance is a concern in this case). All this determine the choice of nitride fuel as having better overall safety characteristics.

The choice of fuel for lead cooled fast reactors is significantly influenced by lead interaction with metal fuel with generation of uranium and plutonium plumbates. Disadvantages of the metal fuel are high-swelling with burnup that requires deep alloying and high porosity as well as phase changes at relatively low temperatures near the lead working temperatures. Under these circumstances, the choice of the same fuel for BN-1200 looks logical too.

It was also taken into consideration that the Russian Federation has more experience with nitride fuel operation and its technology is more developed than for metal fuel. In particular, one can count on 18 years of operation in BR-10, BORA-BORA experiment in BOR-60, where 12.1% h.a. burnup was achieved.

For large power reactors the nitride fuel allows to implement the advantages of CBR~1 reactor cores and suitable fuel cycle: low reactivity margin for fuel burnup, self-sufficient fuel source, lack of need to separate uranium and plutonium, and the required feedback parameters that define reactor safety (reactivity coefficients and effects).

5.2. Reactor testing of nitride fuel in BN-600 and research reactors

A Comprehensive Programme of Numerical and Experimental Research (CPNER) [11] and validation of working capacity of mixed U-Pu nitride (MNIT) fuel is performed for BN-1200 and BREST-OD-300 reactors within the frameworks of PRORYV Project. The goal of CPNER is the validation of working capacity of fuel rods, ensuring the stability of parameters reproduction and required quality of MNIT fuel, fuel assemblies and experimental fuel assembly production technology. At the current stage, CPNER has a goal to verify the initial stage of operation: for BREST-OD-300 maximal fuel burnup of ~6%, maximal damaging dose up to ~85 dpa, for BN-1200 maximal fuel burnup of ~7.5%, maximal damaging dose up to ~95 dpa.

The programme includes improvement of production technology, composition and structure of the MNIT fuel, pre-reactor research of MNIT properties, reactor tests of fuel rods in research reactors (MIR, BOR-60) and commercial reactor BN-600, post-irradiation research of all experimental assemblies. Reactor research is accompanied by pre-test analysis using fuel behaviour codes.

In order to study fuel behaviour produced by carbothermal synthesis, JSC “VNIINM” has fabricated experimental fuel rods that are being tested in BOR-60 reactor. Fuel pellet parameters are as follows: density from 12.0 g/cm^3 to 13.0 g/cm^3, plutonium content from 12%

to 20%, O_2 content <0.15%, C content <0.15%, pellet diameter from 5.9 to 10.2 mm depending on fuel rod type. Total number of fuel rods manufactured – 65.

Lab technology of mixed nitride fuel with the carbothermal synthesis of initial powders developed in JSC “VNIINM” is implemented on a large scale at JSC “SHK” in Seversk, where the capability to manufacture full-scale experimental fuel assemblies for testing in BN-600 is available. This technology was used to manufacture fuel pellets for fuel rods of all Experimental Fuel Assemblies (EFA) that are being irradiated in BN-600 core. A total of 500 fuel rods were manufactured.

Fifteen large-scale FAs were loaded into BN-600. Irradiation of 7 EFAs has been finished by the fall of 2016 with max. burnup of 7.4% h.a. All fuel rods are intact. Nine EFAs of the detachable type with 7 fuel rods in each were placed into BOR-60 reactor for irradiation.

An irradiation device consisting of 7 fuel rods, (6 of which are equipped with sensors for measurement of gaseous fission products pressure under the cladding, fuel rod extension and fuel temperature), is loaded into the MIR research reactor. According to the program, the post-irradiation study will be completed in 2017.

Post-irradiation examination was conducted on BOR-60 fuel rods at maximum burnup of 1.3% h.a. and 3.2% h.a., irradiated with EFA-1 experimental fuel assembly, and CEFA-1 combined experimental fuel assembly of BN-600 at 5.5% h.a. burnup. The obtained results enable the verification of fuel codes and justification of possible extension of assumed service life of experimental fuel assemblies in BN-600 reactor.

6. DESIGN OF BN-1200 COMMERCIAL SODIUM COOLED REACTOR

The BN-1200 design developed within the framework of the PRORYV project makes full use of Russian experience in development and operation of BN-350, BN-600 and BN-800 reactors and was aimed at ensuring its competitiveness with the best designs of thermal neutron reactors. The list of verified and tested design approaches includes:

- Integral primary circuit layout (Fig. 10);
- Three-circuit reactor unit scheme; and
- A variety of equipment technical solutions.

However innovative solutions enabling new safety qualities and CAPEX reduction are of the most interest:

- Decreased power density in the reactor core (from 450 down to 230 kW/m^3) with the CBR~1 and using nitride fuel;
- Two-three times longer fuel campaign and two times longer refuelling interval;
- Complete integration of all sodium systems of the primary circuit in the reactor vessel;
- Emergency system for decay heat removal from the primary circuit which is built in the reactor vessel;
- System of passive reactivity feedback based on the thermal principle;
- Pressure vessel steam generator;
- Refuelling system without accumulation drums, flushing chamber and spent FA drums; and
- Fewer auxiliary systems.

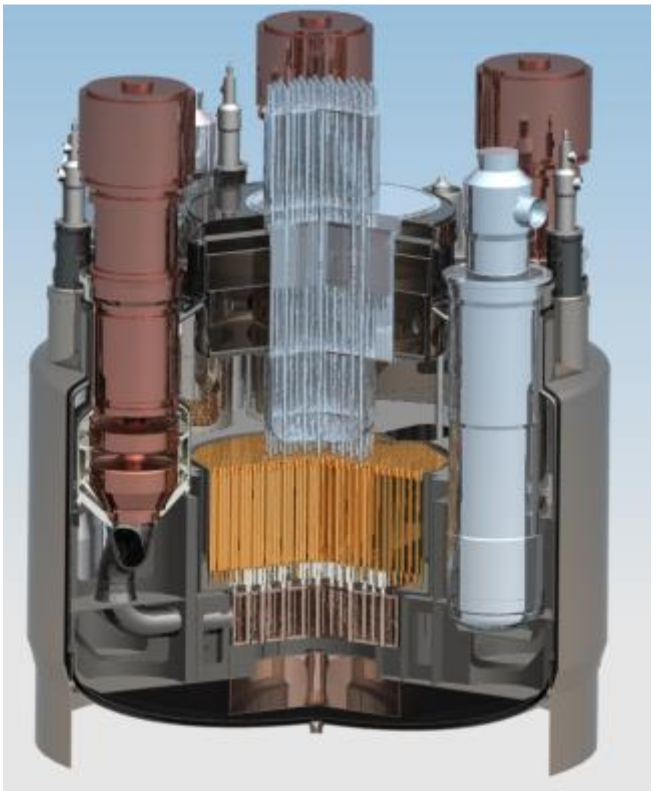


FIG. 10. BN-1200 reactor layout.

Results obtained in the course of design implementation confirm the progress achieved in CAPEX reduction and the technical and economic performance parameters of fast reactors approaching the similar indices of thermal reactors.

At the same time, the PRORYV project is aimed at ensuring the competitiveness not only and, most likely, not predominantly versus thermal reactors, but rather versus alternative power sources including Combined Cycle Power Plants (CCPP) and renewable sources of energy.

The requirements that follow from such a formulation of the objective are detailed in the following section as applied to the BN-1200 fast reactor of 1200 MW(e) power.

7. NEW GENERATION COMPUTATION SOFTWARE FOR VALIDATION OF DESIGN APPROACHES AND SAFETY OF NPPS WITH BREST-OD-300 AND BN-1200 IN CNFC

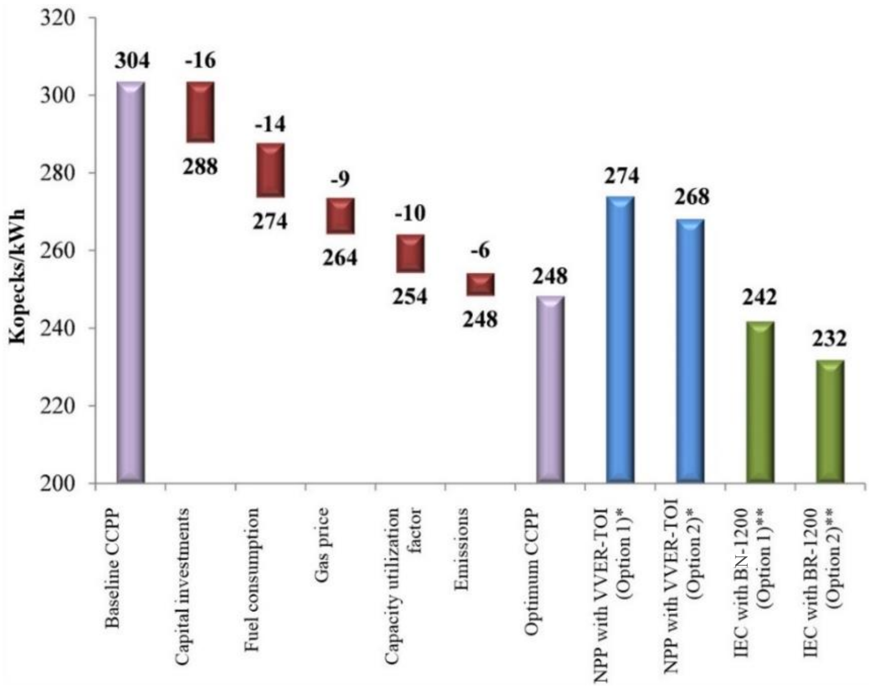
A topic devoted to developing and using new generation computation software for validation and justification of design solutions is opened within PRORYV Project besides the large-scale experimental programme [12, 13]. The list of software codes of new generation includes a full range of software necessary to validate design approaches and safety of NPPs with BREST-OD-300 and BN-1200 reactors, including codes to simulate reactor operation, fission products transport inside the premises of NPP and in the environment, technological processes of CNFC and RAW handling, validated as to both experiments on certain phenomena and the results of BN-600, BN-800, BOR-60 operation in the experimental conditions.

By the end of 2016, the following 18 codes have been developed, verified and validated: neutron physics (MCU-FR, ODETTA, NDP-ACE), thermal hydraulics (HYDRA-IBRAE/LM, LOGOS, CONV-3D), fuel rod behaviour (BERKUT), heat and mass transfer and fission products transport inside premises of NPP (KUPOL-BR), fission products transfer in the environment (Sibilla, ROM, ROUZ), RAW disposal safety validation (GeRa), integral codes for NPP safety validation (EUCLID/V1, EUCLID/V2, SOCRAT-BN/V1, SOCRAT-BN/V2), probabilistic safety analysis (CRISS 5.3), balance of materials and nuclide flows in CNFC (VISART), and three out of them have been already certified by Rostechnadzor, ten others have just entered the certification procedure.

The results obtained with the help of this new generation software have confirmed the high safety level of BREST-OD-300 and BN-1200 designs and allow confirming the main proposals for radiation equivalent RAW disposal.

8. COMPETITIVENESS OF NPP WITH FAST NEUTRON REACTORS IN A CLOSED NUCLEAR FUEL CYCLE

Calculation of Levelized Cost of Energy (LCOE)for fast reactors and Combined Cycle Power Plants (CCPP) under Russian conditions was based on a discount rate of 10%. Fig. 11 shows the results of LCOE calculations based on gradual improvement of CCPP performance versus LCOE of considered NPP units. LCOE values for NPPs are provided for different values of fuel cost.



VVER-TOI – Different price levels for NFC, for BN-1200 – different burnup.

FIG. 11. LCOE of CCPP and NPP in Russian conditions (10% discount rate), kopecks/kW•h.

Table 4 shows the results of the LCOE calculations for competing generation types developed in the Russian Federation with the optimal technical and economic performance for different discount rates.

The following conclusions may be made based on the calculations performed for Russian power generation facilities:

— NPPs with thermal reactors operating in an open Nuclear Fuel Cycle (NFC) cannot guarantee the further efficient competitive development of the nuclear power sector in Russia; and

— Achievement of the established technical and economic performance requirements for industrial energy complexes with BN-1200 will enable maintenance of the Russian nuclear power sector's competitiveness even against CCPPs with an optimum technical and economic performance as well as renewable energy sources.

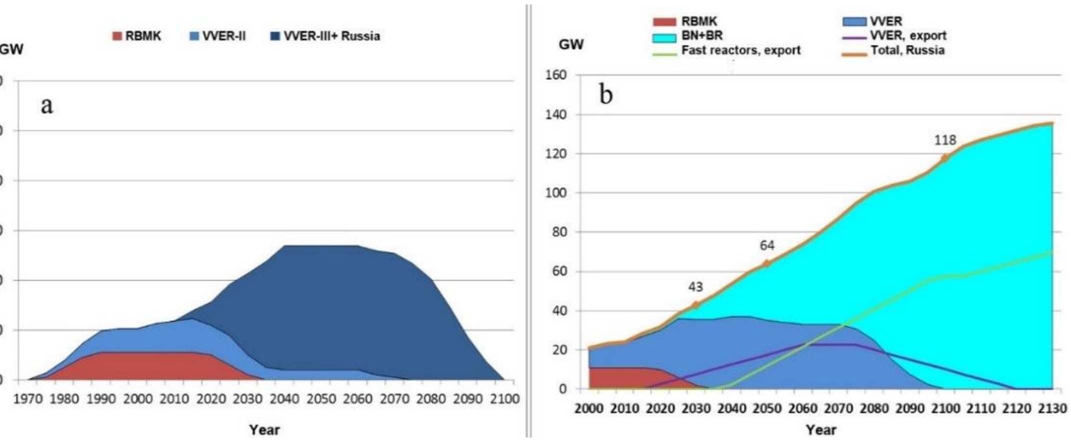
TABLE 4. RESULTS OF LCOE CALCULATIONS FOR COMPETING ENERGY TECHNOLOGIES WITH OPTIMAL TECHNICAL AND ECONOMIC PERFORMANCE, KOPECKS/KW•H

	Discount 10%	Discount 7%	Discount 3%
Solar Power Plant	485.4	284.0	228.8
Wind Power Plant	322.5	189.9	152.9
CCPP	248.3	152.9	136.2
VVER-TOI	268.1	151.8	116.6
BN-1200	231.8	129.2	96.7

9. TWO-COMPONENT NUCLEAR POWER SECTOR AND ITS DEVELOPMENT PROSPECTS

The nuclear power sector has a future only if FR technologies and a closed NFC are successfully mastered (Fig. 12).

Scenario (i) implies that the objective of developing a large-scale nuclear power sector will not be achieved if Water-Water Energetic Reactors (VVERs) with an open Nuclear Fuel Cycle (NFC) is used, so because of uranium resource restriction at the level of 700 kt the introduction of new generating facilities must cease between 2040 and 2045 with power level of the nuclear power sector at <55 GW. In this case, the nuclear power sector will deplete its resources and cease existing by the end of the century. If fast reactors with closed NFC are introduced in due time based on scenario (ii) (first 3-5 units based on fast neutron technology currently in use followed by inherently safe FR with lead coolant), and if installed capacity growth rate until the year 2040 is the same, there will be no resource restrictions for development of nuclear power, and by the end of the century the level of ~120 GW may be achieved with possible future growth.



a – single-component nuclear power based on VVER

b – two-component nuclear power based on VVER→FR

FIG. 12. Possible dynamics of nuclear power development in Russia.

Regardless of the overall Russian nuclear power growth forecast, development of a two-component nuclear power sector should be transient in this century and will end with a transfer to a new technological platform with the domination of inherently safe fast reactors and closed NFC. Duration of this stage should be minimized, if possible, based on the following key aspects:

- Preservation of acceptable nuclear power safety level in general with a significant increase in NPP capacities;
- Uranium resource saving;
- Resolution of the problem with accumulated spent nuclear fuel from thermal reactors; and
- Reduction of system-wide electricity cost (ultimately also in the nuclear power sector).

10. SUMMARY

The results achieved over a relatively small period of time (five years) within the PRORYV project have confirmed technical and technological feasibility of its basic provisions and enable a transition to the practical implementation stage and transfer to a new technological platform of the nuclear power sector based on a closed nuclear fuel cycle at the cusp of the 2030's.

Implementation of the developed design, engineering, process solutions and performance of the scheduled R&D programme at the Pilot Demonstrator Energy Complex (first phase start-up in 2020) will ensure a high probability of appearance of a new prototype of competitive industrial energy complex capable of operating within the framework of the two-component nuclear power sector by 2030.

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4.2. CHINA

Research, development and deployment of fast reactors and related fuel cycle in China*

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* Although a presentation was given, no paper was made available for publication

4.3. FRANCE

Status of the French Fast Reactor Programme

(Extended Abstract)

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Abstract. After an introduction on the current and the foreseen French fuel cycle, the presentation is divided into two parts, the first one considering the different topics addressed by the French Fast Reactor Programme, the second one focusing on the Sodium cooled Fast Reactor Programme.

1. THE RATIONALE FOR A CLOSED CYCLE AND ITS EVOLUTION TOWARDS MORE SUSTAINABILITY

The spent fuel processing and recycling were implemented in France more than 30 years ago. Processing spent fuel allows to salvage recoverable material, whereas the other compounds are considered as “ultimate waste”. All of the plutonium recovered by reprocessing is recycled in mixed uranium-plutonium oxide fuel, called MOX fuel, whereas fission products and minor actinides are incorporated in glass and cast into a metallic canister, then stored in pits pending their final disposal. This policy has a number of advantages in terms of resource saving, plutonium inventory control, reduced ultimate waste (which does not contain large quantities of plutonium) and safe packaging of this waste.

However, multiple plutonium and uranium recycling are not achievable in the current fleet of water reactors. What about the longer term? In order to preserve and to prepare the ability of the next generations to use nuclear power in order to meet their own needs, uranium resource should be preserved and the waste burden should be minimized. The complete closing of plutonium and uranium fuel cycle is an answer to this concern. It implies to perform recurrent and systematic recycling of plutonium and uranium and to operate some reactors to take the best advantage of these recycled materials, within a nuclear fleet.

Fast neutron reactors appear as the most suitable and efficient reactors to reach that aim: they efficiently convert uranium into plutonium and they are able to fission all plutonium isotopes. This appears as the best way to completely use the plutonium, and to open the way to a drastic extension of natural uranium valorization.

2. FAST REACTORS AT THE HEART OF THE CEA RESEARCH

Among the six reactor systems of the Generation IV International Forum (GIF), France pays interest to four of them. France carries out research on those four systems, of course at very different level: Sodium cooled Fast Reactors (SFRs); Very High Temperature Reactors, considered through material requested by such reactor and through hydrogen technologies earlier developed by the CEA; Gas cooled Reactors; Molten Salt Reactors.

The French Gen IV programme focuses on Sodium cooled Fast Reactors, considering both the reactor and the fuel cycle. In fact, France has brought a major contribution to the development of those reactors in the past decades. So far there is a strong connection between the technological maturity of a process and nuclear safety: technological control associated with significant experience feedback contributes to the safety level of a system. Among the Gen IV fast reactor systems, only the SFR has a sufficient knowledge base in France to meet the technical and operational expectations of Gen IV systems in the medium run. However, the French fast reactor programme also includes an active survey and some specific studies on other Gen IV systems, through the European project, through the tasks of the GIF and specific cooperation agreements.

3. THE ASTRID PROGRAMME

The ASTRID programme consists in the design of a test reactor. Its objective is to prepare for the future and ensure that a Gen IV reactor type achieves a technological maturity for the second half of this century. ASTRID means Advanced Sodium Test Reactor for Industrial Demonstration. Based on the feedback experiences of past Sodium cooled Fast Reactors, ASTRID is designed to demonstrate the relevancy and performances of the technology breakthroughs and of the innovative options at an industrial scale, in particular, in the fields of safety and operability.

With the related R&D facilities, ASTRID should allow to test and qualify innovative safety design options towards the commercial reactor, to qualify different fuels including minor-actinide transmutation, to obtain the necessary data to justify a useful lifetime of 60 years for future SFR and to confirm performances of innovative components and systems in order to optimize the design of future commercial reactors from a technical and economical point of view.

The CEA is the leader of the ASTRID project. A “Design Core Team” has been organized through a set of bilateral collaboration agreements. At the time of the FR17 Conference, fifteen industrial partners have joined the ASTRID project.

The role of the CEA involves the operational management by a project team also in charge of the industrial architecture, the management of most of the R&D work and qualification of the technical options, the assessment of studies carried out by its industrial partners in charge of technical work packages and the direct management of the core work package.

The CEA has set up bilateral partnerships with key industry players, in France and abroad. The partners provide both technical and financial support. Support to the CEA Project Owner Team is provided by EDF. The design is organized into engineering batches AREVA NP, now Framatome, shares the design of the nuclear island with a consortium of Japanese partners JAEA, Mitsubishi Heavy Industry and Mitsubishi Fast Breeder Reactors. General Electric is in charge of the power conversion system. Bouygues carries out studies on innovation for the civil engineering. NOX is in charge of the balance of plant, and SEIV of the post-irradiation experiments hot cell. Innovative developments are carried out by Toshiba (sodium electromagnetic pumps), Onet Technologies (inspection machines), Technetics (tightness seals), Velan (sodium valves), CNIM (manufacturing processes) and Airbus Safran Launchers (reliability methods).

About 600 people currently work on the ASTRID project, 60% of which belonging to the industrial partners. Such partnerships enable the CEA to concentrate on the ASTRID pre-conceptual design by implicating key industrial players whose experience and skills in their respective fields are very efficient. The association of different industrial partners offers several advantages: fostering innovation, ensuring that the industrial issues are covered as early as the design phase while providing a source of funding for the pre-conceptual design phases since the partners partially finance the project.

The main international cooperation on the ASTRID project is the cooperation with Japan. Both Japan and France have developed technologies for SFRs for several decades. Collaborative R&D arrangements exist for a long time between the two countries. A common will to cooperate on the ASTRID project emerged in 2013, leading to a “General Arrangement” signed in May 2014 at the government level, followed by an "implementing arrangement" signed by the Japan Atomic Energy Agency (JAEA), Mitsubishi Heavy Industry (MHI), its subsidiary Mitsubishi Fast Breeder Reactors (MFBR), AREVA and the CEA. It shapes the principles and the governance of the R&D and design activities, with 29 "Task-sheets" describing the technical programme, the deliverables, deadlines and input data. In the design field, Japanese team contributes directly to the ASTRID basic design on three topics: active decay heat removal system, control rod system and seismic isolation systems for the reactor building. The R&D field is wide. Major topics are a severe accident with the simulation code and an experimental programme for its validation, R&D on fuel and core materials and reactor technology including instrumentation and components technology, reactor materials and thermal-hydraulics. After nearly three years of common work, it has been shared that the collaboration remains efficient and fruitful and discussion continues to enlarge the shared scope, in particular in the design field.

Since 2007, the CEA has developed a panel of international partnerships in order to share, to reinforce and to spread its R&D efforts. The first framework for those developments has been the European initiatives, among which the European Sustainable Nuclear Industrial Initiative (ESNI), promoting and facilitating efforts of the different partners. Several European SFR dedicated projects have been launched, including recently the Inspyre project on MOX fuel licensing. Beside this quite large network, ASTRID qualification needs to widen R&D collaborations. From 2014 on, CEA has set up new bilateral agreements, called ARDECo framework for ASTRID R&D European Cooperation, abbreviated for ASTRID R&D European Cooperation. This involves many partners in the UK, Germany, Sweden and Switzerland.

Beyond this European R&D network, there are other existing or possible international cooperation: with the Russian Federation on many technical topics and on the use of R&D facilities, with India on safety items and severe accidents, with the US, that performed an evaluation on the core characteristic, and which R&D facilities might be further used, with Kazakhstan on an experimental programme performed in IGR. Several items are discussed with China. The use of Korean R&D platform is also under discussion.

Considering the ASTRID project itself, in December 2015, the ASTRID project met the milestone of the end of the conceptual design. The CEA delivered two synthesis files to the French Government and got the authorization to proceed to the basic design phase until the end of 2019.

The main technical options are discussed here. The thermal reactor power is 1500 MW(th), corresponding to 600 MW(e). This choice results from a balance between industrial demonstration and the cost of the project. ASTRID is a pool type reactor with intermediate sodium circuit. The pool concept has intrinsic advantages regarding safety criteria, with high thermal inertia, guarantee of the inventory in primary sodium. The core concept aims at improving safety in the event of a total loss of coolant accident. The objective is to prevent sodium boiling by implementing a “low void effect core” that maximizes neutron leakage from the core in the event of an accident and thereby reduces the core reactivity in case of sodium temperature increase. The reference fuel for the ASTRID core is mixed oxide $(U, Pu)O_2$. France has a significant experience feedback available, acquired for more than forty years based on experimental programmes and monitoring programmes carried out in Rapsodie, Phenix and Superphenix. Regarding severe accidents, the design integrates a core catcher placed at the bottom of the reactor vessel in order to spread the corium out and cool it. Reactivity is mastered by core spreading and by addition of absorbing materials if necessary. Cooling is mastered by sodium natural convection around the core catcher. The corresponding R&D focuses on corium behaviour and sodium-corium interaction. Decay heat removal systems benefit from the significant boiling margin of sodium in normal operation (more than 300°C) together with a high thermal inertia of the primary system. Those systems are based on natural convection, which allows the use of systems operating in passive mode. As for the cold source, there is a diversification, some systems using air and other water. Fuel handling is designed with a combination of internal and external storage in order to increase the availability rate.

Two options are considered for the power conversion system. During the conceptual design phase, the reference configuration was a steam water power conversion system. This mature option benefits from a large experience and tens of unit operating years. Steam generator design improvement allows to reduce the risk of sodium water reaction occurrence and to mitigate its consequences in case of a hypothetic violent reaction. Another way of dealing with this reaction is to avoid water in the power conversion system. That is why the second option of power conversion system is considered; in this very innovative option, the water sodium reaction risk is inherently eliminated by replacing water with an inert gas. No showstopper has been identified so far. The direction remains to increase the maturity level of the Gas PCS configuration while keeping the two options open.

The general layout is designed with the reference site for the design. The reactor and the safety buildings are located on the rock. Nevertheless, the main buildings are designed to lay on seismic isolation.

In addition to the available experimental platforms in other countries, the French Fast Reactor Programme includes several investments in R&D facilities. The existing platforms are also used to carry out experimental validation of new concept.

4. CONCLUSION

Since nuclear energy is a well proven source of large baseload electricity, with no greenhouse gas emissions, it will remain one of the pillars of the low carbon energy mix in France.

The closed fuel cycle associated with Fast Neutron Reactor (FNR) will lead to drastic improvement in uranium resources management and important reduction in footprint and radiotoxicity of the final wastes.

In this context, the French programme on Gen IV is based on the ASTRID programme on one hand and on an active survey on other Gen IV fast and thermal neutrons systems on the other hand.

The basic design phase of the ASTRID project is ongoing until 2019. Schedule and organization for next phases are under preparation with the French government in the framework of the Japan partnership and with our industrial partners.

4.4. INDIA

Indian Fast Reactor Programme: Status and R&D Achievements

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Abstract. Fast Breeder Reactors form the second stage of India's three stage Nuclear Power Programme based on the domestic nuclear resources. Indira Gandhi Centre for Atomic Research (IGCAR) is primarily dedicated to the broad-based R&D of sodium cooled fast reactors, fuel cycle and associated technologies.

India has been operating a Fast Breeder Test Reactor (FBTR) since 1985, fuelled with a unique plutonium rich mixed carbide fuel (70% PuC + 30% UC). It has so far completed 24 irradiation campaigns in its successful operation over thirty years. Fuels of all types viz., carbide, oxide, as well as metal fuels (both binary and ternary), are currently under irradiation. FBTR has served as a test bed for various experiments, fuel and structural material irradiation, isotope generation programs. The mixed carbide fuel has demonstrated a record burnup of 165 GW·d/t and it has been operated at 400 W/cm peak LHR and at higher operating temperatures. Currently, a 500 MW(e) Prototype Fast Breeder Reactor (PFBR) designed and developed by IGCAR, is in an advanced stage of commissioning. The design of PFBR incorporates several state-of-the-art features and is foreseen as an industrial scale techno-economic viability demonstrator for India's FBR program. IGCAR is presently engaged in the design of 600 MW(e) oxide fuelled FBRs by incorporating many advanced features.

CORAL (Compact Reprocessing of Advanced fuels in Lead cell) facility has reprocessed spent fuel discharged from FBTR with burnup up to 155 GW·d/t and adequate decontamination has been demonstrated. Currently, a Demonstration Fast Reactor Fuel Reprocessing Plant (DFRP) is being established to process both MOX and mixed carbide fuels. A dedicated co-located Fast Reactor Fuel Cycle Facility (FRFCF) for PFBR is under construction. For the future, IGCAR has initiated development programme on metallic fuel. Demonstration of fuel fabrication and pyroprocessing / aqueous technologies for metal fuels on an engineering scale is being pursued.

The R&D areas address all domains of fast reactor science and technology, including sodium technology, safety, materials development, fuel cycle, chemistry, sensors, advanced instrumentation and inspection. This paper presents an overview of the broad-based R&D carried out by IGCAR in the domain of reactor technology, fuel cycle technology, materials development, basic sciences in support of fast reactor program, fuel chemistry, sodium technology, engineering development etc.

Key Words: Indian FBR program, FBTR, Future FBR, CORAL, FRFCF, R&D

1. INTRODUCTION

India's energy requirements continue to grow in line with the industrial growth. Accordingly, Department of Atomic Energy (DAE) has evolved a plan for a significant nuclear capacity addition. DAE has adopted a three phase programme employing its modest uranium reserves and large thorium reserves. The first phase is marked by Pressurized Heavy Water Reactors (PHWRs) and the second phase comprises Sodium cooled Fast Reactors (SFRs). IGCAR is established as the nodal centre for development of fast breeder reactor technology. Currently, a test reactor called Fast Breed Test Reactor (FBTR) and a reprocessing facility are in operation. Based on a comprehensive R&D program, PFBR has been designed which is under commissioning and the co-located FRFCF is under construction. It is continuing with the R&D in all areas related to the FBR technology including the fuel cycle. This paper gives a brief overview of the FBR programme and the IGCAR organization and few major recent R&D are briefly brought out.

2. EVOLUTION OF FBR AND FUEL CYCLE PROGRAMME IN INDIA

Fast Breeder Reactor (FBR) programme in India has started with the construction a 40 MW(th) sodium cooled mixed carbide fuelled Fast Breeder Test Reactor (FBTR) in 1971 and successfully operated for 32 years. Currently, 500 MW(e), MOX fuelled Prototype Fast Breeder Reactor (PFBR) is in an advanced stage of commissioning by Bhartiya Nabhikiya Vidyut Nigam Limited (BHAVINI) [1]. Subsequent to PFBR, six units of MOX fuelled Sodium cooled Fast Reactors (SFR) would be constructed on a commercial basis and two units would be deployed at Kalpakkam. This would be followed by metallic fuelled reactors. Based on the valuable experience gained from design, manufacture, erection and safety review of PFBR, a preliminary conceptual design for the future 600 MW(e) FBR (FBR1/2) has been completed. Since the remaining projected life of FBTR is about five EFPY (Effective Full Power Years), an initiative has been undertaken to work on and evolve a new high flux fast spectrum test reactor (tentatively called FBTR-2), of 100 MW(th) capacity. Preliminary design of its core has been achieved.

With reference to fuel reprocessing, adequate experience has been accumulated in the aqueous reprocessing technology through the reprocessing of mixed carbide fuel and the MOX fuel irradiated and discharged from FBTR. For the metallic fuel, development of pyro-processing technology is currently underway. As part of the ongoing studies on the direct electrochemical conversion of oxide to metal in molten salt, studies were carried out and feasibility to reduce solid ZrO_2 to zirconium under specific experimental conditions was established in lab scale. To gain experience in engineering scale pyroprocessing, an engineering scale R&D facility on the various process steps of pyrochemical flow sheet for alloy fuels is being setup.

3. IGCAR ORGANIZATION

Indira Gandhi Centre for Atomic Research (IGCAR), the second largest establishment of the Department of Atomic Energy, was set up in the southern part of India at Kalpakkam, 80 km south of Chennai (MADRAS), in 1971 as Reactor Research Centre and rechristened as IGCAR in 1985. The main objective of the centre is to conduct broad-based multidisciplinary programme of scientific research and advanced Engineering, directed towards the development of sodium cooled FBR technology, fuel cycle and associated technologies in India which are being carried out through a host of multidisciplinary laboratories (see Fig. 1). In meeting the objective, a modest beginning was made by constructing FBTR. With the experience and expertise gained by the successful operation of FBTR, the Centre has designed PFBR and extended the technical expertise and support to BHAVINI in the commissioning of the reactor. A 30 kW(th), ^{233}U fuelled Kalpakkam Mini reactor (KAMINI) has been operational for many years for neutron radiography, neutron activation analysis etc.

FIG. 1. IGCAR Mission Areas.

Over the years, the centre has established comprehensive R&D facilities covering the entire spectrum of FBR technology related to Sodium Technology, Reactor Engineering, Reactor Physics, Metallurgy and Materials, Chemistry of Fuels and its materials, Fuel Reprocessing, Reactor Safety, Control and Instrumentation, Computer Applications etc., and has developed a strong base in a variety of disciplines related to this advanced technology. More details about the development of fast breeder technology in India can be obtained from [2]. As a part of efforts for closing the fuel cycle, a dedicated Fast Reactor Fuel Cycle Facility (FRFCF) for PFBR is also coming up close to the PFBR site. Various units of DAE are involved in the FRFCF project which is piloted by IGCAR.

IGCAR utilizes its expertise and resources in enhancing its standing as a leading Centre of research in various branches of basic, applied and engineering sciences that have a bearing on Nuclear Technology like Structural Mechanics, Heat and Mass Transfer, Material Science, Fabrication Processes, Non-Destructive Testing, Chemical sensors, High temperature thermodynamics, Radiation Physics, Computer science etc. Apart from thrust areas related to nuclear technology, the Centre has credentials as a leader of research in various frontier and topical subjects like Quasi crystals, Oxide superconductors, Nano-structures, clusters, SQUID fabrication programs, exopolymers and experimental simulation of condensed matter using colloids etc.

A modern library comprising 62,000 volumes of books, 28,400 back volumes, about 785 journals and 195,000 reports in all disciplines catering to the technical needs of the scientists and engineers. The Central Workshop is fully equipped with sophisticated machines for the fabrication of precision components. The Computer Division houses advanced high-performance computing servers and application packages to meet the computational demands of the users.

4. FAST BREEDER TEST REACTOR (FBTR)

FBTR has completed 31 years of operation and generated valuable experience in various aspects of operation, maintenance and core management. The knowledge gained through successful operation of FBTR has provided vital inputs for the commissioning of fast breeder programme through the construction and commissioning of PFBR. The reactor was very useful in mastering sodium cooled fast reactor technology and testing/validation of advanced fuels, structural materials, instruments and equipment. It has reached the highest power level of 27.3 MW(th) and 6.0 MW(e). Mark-I and Mark-II fuels have achieved a maximum burnup of 165 GW·d/t and 100 GW·d/t respectively. The primary sodium temperature was increased to rated design value at reduced power. Presently the reactor is being used for testing/validation of fuels, structural materials, instruments and equipment through a comprehensive irradiation programme. The irradiation campaigns include long term irradiation of D9 alloy (~29 dpa achieved), low dose irradiation of 304LN and 316LN stainless steels (~5 dpa achieved), Ferro-boron shielding material (2.96 dpa achieved), TRISO-coated ZrO_2 kernels and Nb-1Zr-0.1C disc specimens for HTGR, sodium-bonded metallic fuel pins of U-6Zr & U-Pu-6Zr, Yttria for ^{90}Sr production for therapeutic purpose and testing of high temperature fission chambers for PFBR.

The major upgradation carried out in FBTR includes, construction of flood safe building to accommodate two air cooled emergency diesel sets, installation of supplementary control panel, ramps at entry points to prevent flood water entry, seismic strengthening, reinforcement of URM walls, fire water system upgradation with submersible water pumps, installation of 2×5 m^3 capacity Demineralization (DM) water plant. Post Fukushima retrofits include installation of solar powered LED street lights in strategic locations, augmentation of the diesel-generator (DG) oil storage tank, addition of two 140 kVA mobile diesel generators and improvement in plant security.

5. PROTOTYPE FAST BREEDER REACTOR (PFBR)

Currently, PFBR designed and developed by IGCAR, is in an advanced stage of commissioning. BHAVINI, a fully owned Government company, has the responsibility of construction, commissioning and operation of PFBR. IGCAR is extending the necessary technical expertise and support for the construction, commissioning, safety clearance from regulators and the first criticality. The design of PFBR incorporates several state of the art features and is foreseen as an industrial scale techno-economic viability demonstrator for India’s FBR program. More details on PFBR are covered in many companion papers in this conference.

6. FUTURE FBR DESIGN – APPROACH AND DESIGN OBJECTIVES

Subsequent to PFBR, six units of MOX fuelled Sodium cooled Fast Reactors (SFR) would be deployed for which the detailed design is in progress as mentioned earlier. Based on detailed studies, the power of these reactors has been finalized to be 600 MW(e). These six reactors will be deployed as a twin unit design.

The major design objectives for FBR1/2 are: (i) Improved economy and higher power output with nearly same reactor assembly size; (ii) Enhanced safety aiming Gen III+; iii Sodium void reactivity < 1 $; (iv) Breeding ratio as high as possible and optimum fuel inventory; (v) Possible higher operating temperatures towards higher efficiency; (vi) Optimum number of heat transport systems and components; (vii) Maximum utilization of the manufacturing technology established for PFBR; (viii) Reduction of specific capital cost and construction time; (ix) Incorporation of inherent and/or passive safety features to terminate severe accidents; and (xi) elimination of the need for offsite public evacuation.

The approach adopted in the design to achieve the objectives are: (i) Retaining the standard design options which have been validated in the design of PFBR; (ii) Optimization and simplification of component design leading to reduction in specific capital cost; (iii) Optimum number of primary heat transport systems; (iv) Improvement in the design based on current state of art manufacturing technologies; (v) Advanced design concepts and features towards economy and safety; (vi) Highly reliable engineered safety systems; (vii) Incorporation of inherent and/or passive features to terminate severe accidents; (viii) Use of alternative materials for high-performance and economy; (ix) Increased burnup; (x) Twin units layout with sharing of facilities without compromising safety; (xi) Higher thermodynamic efficiency through higher plant operating temperatures; (xii) Design for higher plant life; (xiii) Design features facilitating reduced construction time and parallel construction; (xiv) Reduced fuel cycle cost through higher burnup and lower throughput. With this approach, the reactor is being designed for a power of 600 MW(e) with MOX fuelled homogeneous core having higher breeding ratio, sodium void reactivity less than $ 1. More details about the design are discussed in a companion paper [3].

7. R&D ACHIEVEMENTS

A comprehensive R&D programme has been undertaken in IGCAR which has provided input for evolving the design of 500 MW(e) PFBR. The R&D programme is continuing to address the emerging areas for the design of future reactors and reprocessing technology. A few major important R&D activities carried out in the recent period are discussed briefly in the following sections.

7.1. Major Design Aspects of FBR1/2

Several core designs were studied and a homogeneous core design has been selected which has two enrichment zones of mixed oxide (PuO_2-UO_2) fuel, followed by blanket and reflector, in-vessel storage and Ferro-Boron (Fe-B) sub-assemblies (outer shield). With the final core design, the sodium void coefficient is reduced to $ 0.9 and the breeding ratio achieved is 1.11[4]. Based on the analysis it is seen that the increased burnup of about 150 GW·d/t is possible by using ferritic steel as the wrapper material.

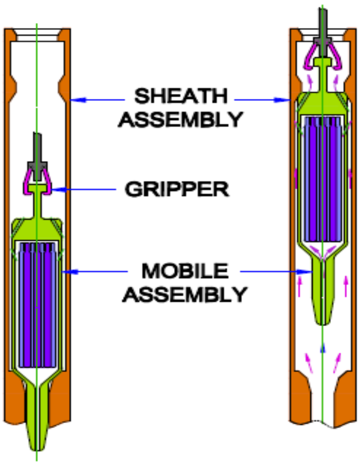

FIG. 2. Hydraulically suspended absorber rod

For higher reliability of Shutdown Systems (SDS) compared to PFBR, two types of SDS are provided in the design consisting of nine Control and Safety Rods (CSRs), three Hydraulically Suspended Absorber Rods (HSARs) (see Fig. 2) and three Diverse Safety Rods (DSRs). To improve the reliability of SDS further, elaborate R&D has been carried out on (i) stroke limiting device in CSRDM and (ii) the temperature sensitive magnetic switches in DSRDMs.

To enhance the safety and economy, the SG length has been increased to 30 m. The availability and reliability of Decay Heat Removal (DHR) systems will be enhanced during fuel handling, in-service inspection, design basis events, design extension and post accident conditions. The SGDHR is envisaged to have four DHR circuits (two with forced circulation and two with natural circulation), each with 10 MW(th) heat removal capacity [5]. In addition, to meet the DHR requirement during post accident situations, the concept of vessel cooling is being developed. The core catcher is designed to take care of whole core melt down with B_4C spikes introduced to prevent re-criticality. To take care of high temperature molten corium attack, the core catcher materials/coatings are under development.

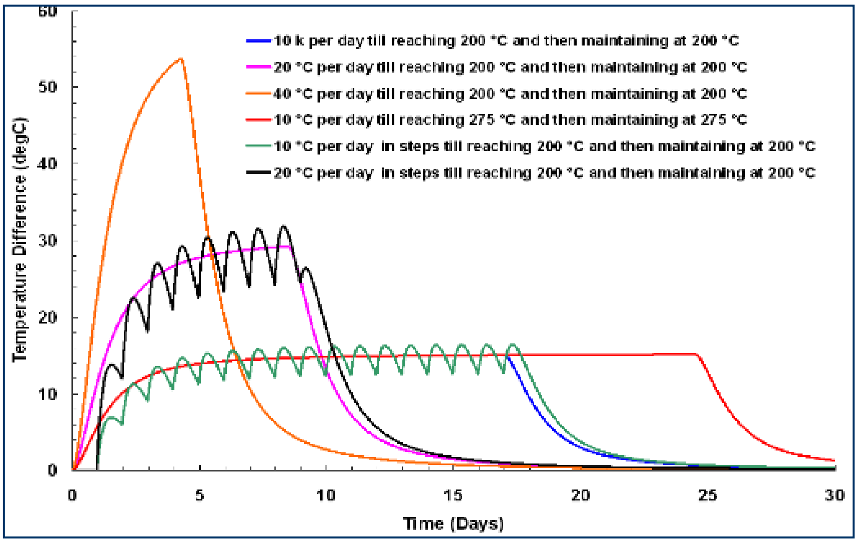

FIG. 3. Evolution of temperature difference between grid plate and inner vessel during preheating.

Several thermal hydraulic studies have been carried out to optimize the reactor design and also to resolve PFBR commissioning issues. Few recent studies include PFBR pre-heating studies for primary and secondary systems which finalized heating rate of 10°C/day (see Fig. 3), FBR1/2 SG inlet flow distribution studies to select suitable flow distribution devices viz., porous plate and porous shell, flow zoning studies for IHX secondary sodium using CFD models to get uniform tube outlet temperatures. enhance flow through outer rows of tubes.

7.2. Experimental Studies and facilities

Seismic studies: Several shake table tests have been conducted to study dynamic behaviour of the piping systems and components during earthquake conditions. The proposed double acting pneumatic operated butterfly valves for pre-heating and emergency core cooling circuit of FBTR were seismically qualified under review level earthquake conditions (see Fig. 4). After testing, the structural integrity and functional operability were checked.

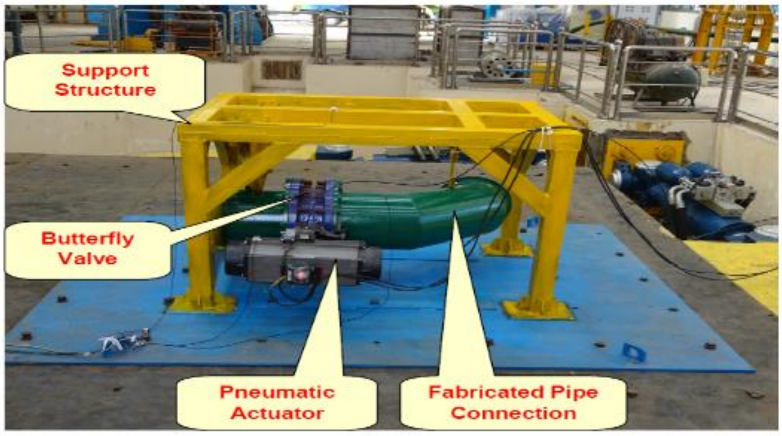

FIG. 4. Shake Table Experimental Setup.

Gas entrainment studies: Studies were carried out in a 5/8th scale 90° sector water model of primary circuit of future FBRs. Gas entrainment mitigation devices have been developed using Computational Fluid Dynamics (CFD) studies (see Fig. 5). Performance of these devices have been tested in this model. Free surface velocity was reduced by combination of horizontal baffle plate attached to inner vessel baffle plate and graded porosity skirt.

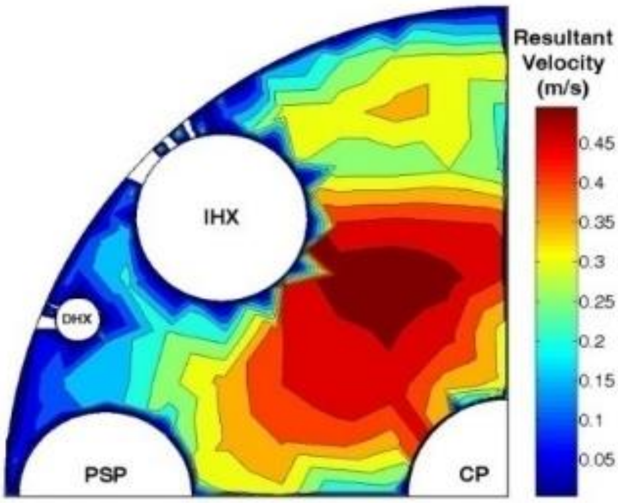

FIG. 5. CFD Result of 5/8 Scale Model.

Source assembly antimony pin wetting: Experimental investigation of flow pattern and wetting behaviour inside source assembly has been carried out. In this experiment, visual inspection and characterization of locked gas bubbles in the annular space between the pins at different experimental condition were carried out. It is seen that few gas bubbles are locked inside the annular space in water experiment inside glass tube. Wetting experiment in sodium demonstrated that there is no wetting problem.

FIG. 6. Photograph of ICT and submersible ALIP.

Integrated cold trap: A full scale model of Integrated Cold Trap (ICT) (see Fig. 6) was manufactured and conducted performance testing in hot sodium pool. Purification of 16 tonnes of sodium was experimentally demonstrated. Performance of submersible Annular Linear Induction Pump (ALIP) was satisfactory. The overall performance of newly developed ICT is found to be in line with the design expectations.

DSR Drop time measurement: Simulation of reactor primary circuit with 91 subassemblies including three Diverse Safety Rods (DSR) and three Control Safety Rods (CSR) were carried out in water facility. Acoustic signals of simultaneously dropped DSR were obtained using accelerometer fixed on wave guide to obtain DSR drop time measurement.

7.3. Recent Post Irradiation Examination Studies

As mentioned earlier several irradiation tests are being carried out in FBTR and the Post Irradiation Examination (PIE) of all the test specimens is in progress. PIE carried out on the sphere-pac fuel pins to assess the Beginning Of Life (BOL) behaviour has shown (see Fig. 7) that the different size fraction micro-spheres are still discernible, indicating that gross re-structuring had not taken place though sintering of the micro-spheres had initiated along with appearance of porosities in the larger particles, especially at the centre of the fuel column. This study has given valuable inputs for the planning of further irradiation experiments with enhanced linear power ratings.

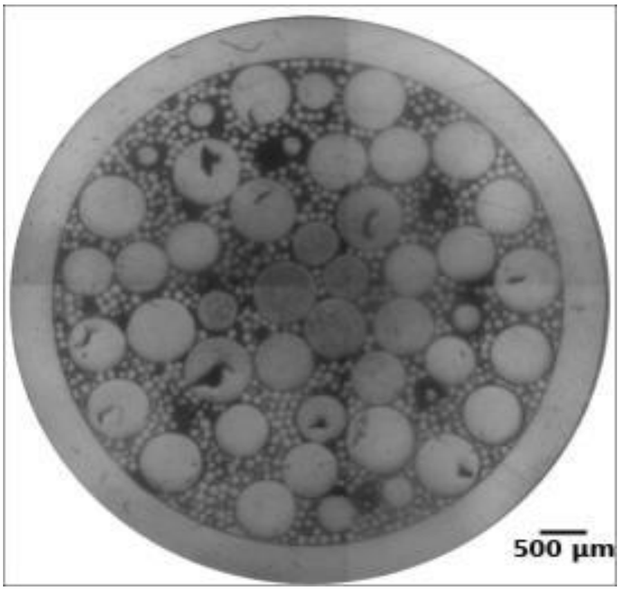

FIG. 7. Sphere-pac fuel pin cross-section at the peak power location.

7.4. Materials Development

Materials development initiatives have been taken to support cost reduction and improved safety in future FBRs. R&D has been initiated in development of new materials for cladding, shielding and structural materials.

The out of pile experiments conducted on ferro-boron gave encouraging results and hence, several irradiation tests are being carried out in FBTR. For improving structural materials, a roadmap leading to a target burnup of 200 GW·d/t with metallic fuel in T91 ferritic martensitic cladding and T9 wrapper is being implemented in a stagewise manner. A phosphorous-containing austenitic stainless steel designated as IFAC-1 has been developed to have enhanced void swelling resistance. To leverage the high void swelling resistance of ferritic steels for next generation cladding applications, thin-walled cladding tubes of oxide dispersion strengthened 9% chromium ferritic-martensitic steel have been manufactured and shown to meet design requirements. Further studies are on-going to produce cladding tubes with higher (>12%) chromium that would exhibit better compatibility with existing fuel reprocessing flow-sheets.

Towards improving structural alloys for out of core components, the effect of low dose neutron irradiation on mechanical properties and microstructural evolution in austenitic steels for permanent core structures was examined through PIE of SS316L(N) and SS304L(N) specimen irradiated in FBTR. The SS304L(N) exhibited a higher rate of hardening and correspondingly a lower residual ductility compared to SS316L(N). It is also seen that 0.14%N is optimal for enhanced creep strength, low-cycle fatigue and weldability. The corresponding welding electrodes required for fabrication of components have also been developed. Improvements in steam generator materials is also envisaged for future FBRs. Welding being the primary fabrication route for steam generators, factors leading to weld failures have been studied. Controlled boron addition to the extent of ~100ppm has been shown to have a beneficial effect along with control of nitrogen.

7.5. Sensors Development

Sensors for measuring hydrogen in liquid sodium and in cover gas (argon) have been developed (see Fig. 8.). The Electrochemical Hydrogen Meter (ECHM) which is useful in identifying steam leak into sodium during normal reactor operating conditions works on the principle of a galvanic cell that generates an Electromagnetic Field (EMF) proportional to the concentration of hydrogen in liquid sodium. The development and implementation of this sensor system, comprising sensor probe, and the allied instrumentation has been successfully accomplished in-house. About ten numbers of this sensor would be installed in PFBR. In order to detect hydrogen in the argon cover gas to monitor steam leak into the steam generator section, a Hydrogen in Argon Detector (HAD) has been developed. The sensor system comprises a Thermal Conductivity Detector (TCD), nickel coil assembly and a gas manifold. The nickel coil serves as a semi-permeable membrane through which hydrogen alone diffuses and is detected by using a TCD. Four of these sensors would be deployed in PFBR.

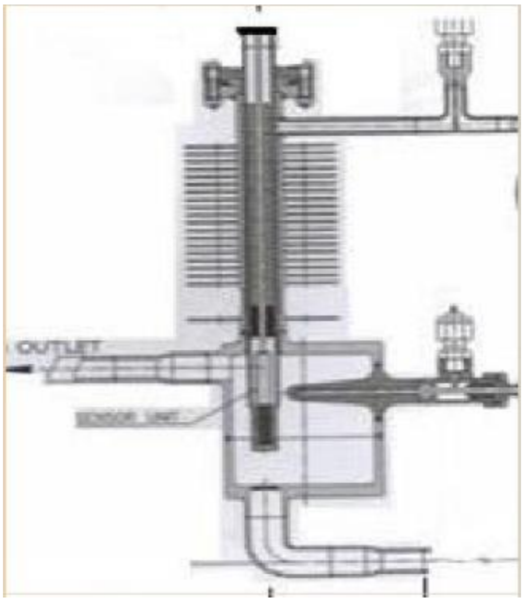

FIG. 8. Eelectrochemical Hhydrogen Mmeter (ECHM) housing.

7.6. In-Service Monitoring and Inspection

Extensive R&D is being carried out to improve in-service monitoring and inspection of structures and components. In order to inspect inaccessible welds, especially in core-support structures, a novel ultrasonic technique using high-frequency guided waves has been developed and optimized for one such application (see Fig. 9). Experiments showed that 20% wall thickness defects can be detected with this technique. A dedicated ultrasonic analysis and imaging software (IGUANI) has been developed for weld mapping in C-scan imaging mode. Robotic devices combining automated motion control with NDE data acquisition system have been developed. One such device, DISHA, has been optimized for the ultrasonic inspection of dissimilar welds (stainless steel to carbon steel between PFBR main vessel and roof slab shell), where this imaging technique has been implemented and demonstrated.

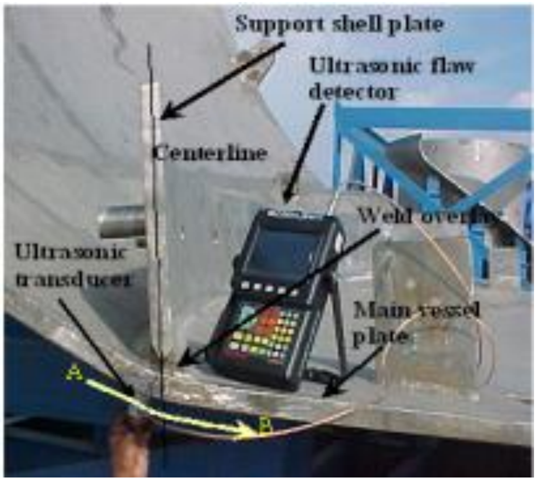

FIG. 9. MV-CSS mock-up sector.

For inspecting 23 m long - 540 tubes steam generator, a Remote Field Eddy Current (RFEC) instrument was developed in-house, incorporating a 30 m flexible RFEC probe. This was delivered down each SG tube in sequence by an in-house developed PFBR Steam Generator Inspection System (PSGIS) system to successfully complete the regulatory authority mandated inspection.

7.7. Instrumentation and Control (I&C) and Electronics Development

IGCAR has successfully developed in-house, the complete Instrumentation and Control (I&C) system for PFBR which includes Hardware Systems, Computer based I&C systems, Switch Over Logic System, Remote Terminal Units and Distributed Digital Control System. All the hardware systems after successful development and functional testing have been qualified for environmental, EMI/EMC and Seismic tests as per international standards. The complete software has been developed in-house and independently verified and validated as per regulatory guidelines. These systems are utilized for monitoring and controlling different plant parameters pertaining to sodium, argon, nitrogen and other auxiliary systems and for fuel handling operations. All these computer based I&C systems are being commissioned at the PFBR site.

R&D activities are being carried out towards building I&C systems based on new technologies to address the obsolescence of components and systems. Development of Advanced Distributed Digital Control System, Diversified Real Time Control Systems, Safe and Secure PLC system, Radar Level Probe, Position Drive System for Failed Fuel Localization Module (FFLM), Post Accident Monitoring Systems, Condition Monitoring of rotating equipment, Severe Accident Monitoring System, Reliability Prediction of Electronic Components and Software Quality Assurance, are the step towards meeting the future challenge of Fast Breeder Reactor’s requirements.

R&D is also being carried out in diversified areas such as Microelectromechanical Systems (MEMS) pressure sensor, fiber optic based sodium leak detection system, hardware trojans, microprocessor based shutdown systems, high temperature fission chambers for neutron flux monitoring, thermocouple probes with three thermocouples and leak tight penetration assemblies for instrumentation cables and fiber optic cables.

Several activities are being carried out towards development of Full Scope Replica Type Operator Training Simulator for PFBR, 3D modelling, animation and visualization of the FBR subsystems. The high-performance scientific computing facility at Computer Division comprise of HPC Clusters, Compute-intensive Servers, Graphic-intensive Workstations, high-end peripherals and advanced application software required to meet the computing requirements of engineers and scientist of IGCAR. A web-based Nuclear Knowledge Management system with advanced features to acquire, store, share and utilize the organizational knowledge has been developed for the fast reactors and associated domains.

7.8. Reprocessing Development

Fast Reactor Fuel Reprocessing: Compact Reprocessing of Advanced Fuel in Lead cell (CORAL) (see Fig. 10) has successfully reprocessed FBTR spent fuel of various burnups viz., 25, 50, 100 and 155 GW·d/t and continues to operate. Demonstration Fast reactor fuel Reprocessing Plant (DFRP) is conceived with the objective of regular processing of spent fuel from FBTR and demonstration of reprocessing of the PFBR fuel. Equipment unique to fast reactor reprocessing plants such as sub-assembly dismantling machine, single pin chopper, feed clarification centrifuge and centrifugal extractor are employed in this plant. Automated robot-based sampling system, is also incorporated in DFRP. The construction of DFRP is nearing completion and commissioning activities have started and is progressing in a phased manner. Water runs are completed and acid tri-n-butyl phosphate (TBP) runs will be taken up shortly followed by natural uranium runs.

FIG. 10. CORAL Hot Cell Facility.

Towards process and equipment development, few major R&D activities carried out are: establishment of a 20 stage ejector type mixer settler, solvent extraction with dual scrubbing, study on dissolution kinetics of MOX fuel, development of compact distillation unit for solvent recovery (see Fig. 11), study on removal of dissolved TBP from aqueous streams of PUREX Process by n-Dodecane wash, development and performance evaluation of RFD based fluidic pump (see Fig. 12) for high discharge head application, development of Inline static mixer based pulse column (see Fig. 13) for reprocessing application etc.

FIG. 11. Setup for solvent recovery studies.

FIG.12. RFD based-fluidic pump.

FIG. 13. Inline static mixer based pulse column.

Pyrochemical Reprocessing Development: Development of flow sheet for pyrochemical reprocessing of metal fuels is currently underway at IGCAR (see Fig. 14). In order to gain experience on co-recovery of uranium and plutonium, molten salt electrorefining on U-Pu-Zr alloy in LiCl-KCl-UCl_3 at 773 K is being carried out. Tantalum crucible with U-19Pu-6Zr (wt.%) was taken as anode and LiCl-KCl eutectic containing 4.71 wt.% uranium as electrolyte. Investigations on the melt composition using cyclic voltammetry and determination of percentage recovery of uranium at cathode have also been carried out. In order to gain experience on engineering scale pyroprocessing, an R&D facility with a capacity of 10 kg of U-Zr alloys per batch is being setup which include a High Temperature Electro Refiner.

FIG. 14. Metal fuel pyrochemical processing facility.

7.9. Metal fuel development

Demonstration Facility for Metal Fuel Fabrication: Sodium bonded metallic fuel pins containing U-23wt.%Pu-6wt.%Zr ternary alloy as fuel slug and U-6wt.% Zr as blanket slug in T91 clad are being developed. As part of this endeavor an injection casting equipment was designed, fabricated and installed inside a glove box for fabricating U-Pu-Zr metallic fuels. After successful installation and validation, defect-free fuel grade U-6Zr slugs were fabricated. The as-cast slugs were subjected to chemical, metallurgical and physical characterization. The fuel slugs were then loaded in T91 clad tubes, annular gap is filled with sodium and closed by top end plug and TIG welded. The fuel pins were then subjected to dimensional inspection, HLT and full length 'X' radiography. Wire wrapping using 0.9 mm dia. wire of 316L material and spot welding on the top plug was carried out on the fuel pin. The fuel pins were inserted and assembled in a sub-assembly form for carrying out test irradiation in FBTR (see Fig. 15).

Development of sphere-pac fuel pins: A fuel fabrication facility based on the sol-gel process is being set up at IGCAR. Using MOX fuel microspheres of 780 mm diameter and natural UO_2 microspheres of 115 mm size sphere-pac fuel pins are fabricated by vibro-packing method and test irradiated (1600 MW·d/t) in FBTR [6-11]. PIE has been carried out on the sphere-pac fuel pins to assess the Beginning of Life (BOL) behaviour such as restructuring of the sphere-pac fuel column and column stability. The results obtained from the irradiation test indicated that though significant restructuring of microspheres to form a pellet type structure has not occurred, the onset of necking and sintering of microspheres could be observed. Segregation / relocation of the two sizes of microspheres were not observed.

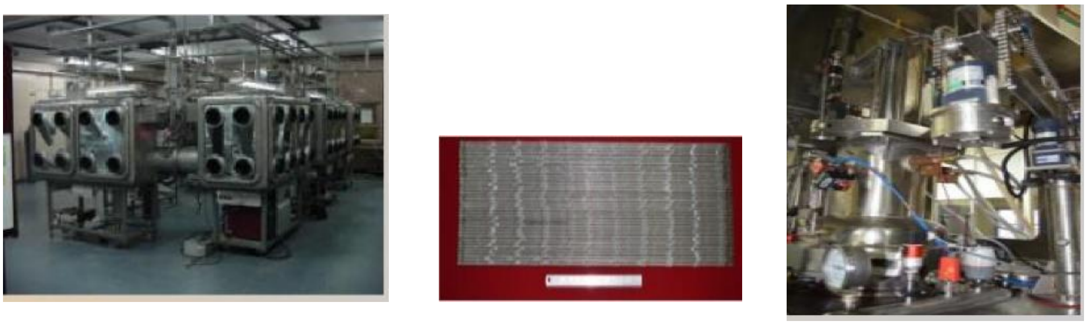

FIG. 15. (a) Metal fuel Fabrication facility. b) injection casting. c) sodium bonded fuel pins.

8. SUMMARY

Fast breeder reactors form the second stage of the Indian nuclear power programme. It is vital for India's energy security and sustainability considering the available nuclear resources in India. Development of FBR technology has started with the establishment of a dedicated research centre called IGCAR and launching of a test reactor called FBTR. With the experience and expertise gained by the successful operation of FBTR, IGCAR has evolved the design of 500 MW(e) Prototype Fast Breeder Reactor (PFBR), which is in advanced stage of commissioning. Based on the valuable experience gained from design, manufacture, erection and safety review of PFBR, IGCAR has arrived at a preliminary conceptual design for the future 600 MW(e) FBR with due considerations to improved economy and enhanced safety. Subsequent to MOX fuelled reactors, metal fuel based reactors are going to be launched. To enable this, metal fuel development is undertaken and to get experience, a test reactor is planned. Towards ensuring higher growth rate, R&D on metal fuel with high breeding potential along with associated fuel cycle technologies is in progress. Towards addressing the technological challenges, comprehensive and challenging R&D activities are taken up in several areas such as reactor physics, advanced shutdown systems with passive features, component testing and development, sensor development, advanced and improved materials development, advanced I&C and electronics, process & equipment development in the domain of reprocessing technologies. India continues to put strong emphasis on the R&D towards building up a substantial fast reactor programme in the future.

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4.5. JAPAN

Current Status and Future View of the Fast Reactor Cycle Technology Development in Japan

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Abstract. The "Fourth Strategic Energy Plan" of Japan was approved by the Cabinet in April 2014. It states that nuclear energy is an important baseload power source as a low carbon and quasi-domestic source even after the TEPCO's Fukushima Dai-ichi Nuclear Power Station (1F) accident, and Japan promotes nuclear fuel cycle in terms of the efficient use of resources and volume reduction and mitigation of degree of harmfulness of high-level radioactive waste and carries out Fast Reactor (FR) cycle R&D for the commercialization, taking advantage of international cooperation.

Japan Atomic Energy Agency (JAEA) is conducting several R&D activities for the commercialization of FR cycle primarily focusing on (i) the reduction in volume and toxic level of radioactive waste and (ii) the improvement of the safety of FRs and utilizing international cooperation with bilateral frameworks such as ASTRID programme with France and multilateral frameworks such as the Generation IV International Forum (GIF). In the nuclear fuel cycle R&D, Small Amount of Reused Fuel Test (SmART) cycle project to conduct a small-scale Minor Actinide (MA) recycling using existing facilities is in progress. Regarding the experimental FR Joyo, JAEA completed the replacement work for the damaged Upper Core Structure (UCS) and it is preparing to make an application for an earlier restart under the new regulatory requirements developed based on lessons learned from the 1F accident.

Meanwhile, the Council on FR Development was established in September 2016 to discuss and prepare a draft paper on policies concerning the future development of FRs in Japan. Based on the result, the Inter-Ministerial Council for Nuclear Power (Inter-Ministerial Council) made a decision on the new policy for FR development in Japan and it states that a strategic roadmap will be compiled in 2018 for the realization of the policy. The Inter-Ministerial Council also decided that Prototype Fast Breeder Reactor Monju will not resume operation as a reactor and will be decommissioned.

Key Words: Fast Reactor (FR) cycle, new policy for FR development, Strategy Roadmap, decision on the decommissioning of Monju

1. INTRODUCTION

In order to secure a quasi-domestic energy source for a long term, Japan, which has few natural resources, started to promote R&D for Fast Reactors (FRs) aiming for the commercialization in the dawning era of the nuclear development. Triggered by the long term Programme for Research, Development and Utilization of Nuclear Energy, developed in 1956 by the Atomic Energy Commission (AEC), a full-scale design study on FRs and related R&D were commenced in around 1963, and the Experimental FR Joyo (Joyo) and the Prototype Fast Breeder Reactor (FBR) Monju achieved first criticality in 1977 and 1994, respectively.

Monju had suffered a sodium leak accident in 1995 followed by an In-Vessel Transfer Machine falling accident but high expectations were placed on its restart both at home and abroad [1]. Since the late 1980s, an R&D project for the realization of a demonstration reactor has been conducted with the cooperation of private entities and the Government, and Japan has solemnly put an effort in accumulating knowledge [2-3].

As it is essential for the development of FR to be carried out together with the development of a nuclear fuel cycle, the promotion of the nuclear fuel cycle has been the basic policy of Japan since the initial stage of the FR development. The 4th Strategic Energy Plan in Japan [4], approved by the Cabinet in April 2014, states that the nuclear fuel cycle contributes to the resolution of the challenge related to disposal of spent fuels, the reduction of the volume and harmfulness of high-level radioactive waste, and effective utilization of resources for mitigating the risks for and the burden on future generations. Japan will make efforts to create the nuclear fuel cycle while taking the past history into consideration and continuing to seek the understanding of relevant municipalities and the international community. It also states that Japan will promote reprocessing and plutonium use in LWRs and R&D of FRs, etc., through international cooperation with the U.S., France, etc. FR further enhances the effect of the reduction of the volume and harmfulness of high-level radioactive waste and effective utilization of resources expected in the nuclear fuel cycle. Moreover, the depth of technologies and human resources cultivated so far will greatly contribute to the formation of technology infrastructures and is the source of the acquisition of cutting edge technologies and international contribution. The significance of Japan's FR development does not change even when the situation has changed recently.

Meanwhile, there have been a variety of changes, such as the formulation of new regulatory requirements, the inauguration of Japan-France cooperation in developing FRs, and Electricity Systems Reform, in the environment surrounding the FR R&D recently, particularly since the TEPCO's Fukushima Dai-ichi Nuclear Power Station (1F) accident in March 2011. In light of the latest situation, the Council on FR Development [5] was established to discuss future approaches of the FR development at a meeting of the Inter-Ministerial Council for Nuclear Power (Inter-Ministerial Council) held in September 2016. The new policy for FR development in Japan [6] and the policy on the Monju [7] were decided in the meeting of the Inter-Ministerial Council held in December 2016.

This paper describes Japan's energy and nuclear policies related to an FR cycle issued since 2013, current status and future view of the FR cycle technology development in Japan

2. JAPAN'S OVERALL ENERGY POLICIES AND NUCLEAR POLICIES RELATED TO AN FR CYCLE

The following are excerpts from Japanese Government's decisions on the overall energy policies and nuclear policies related to an FR cycle issued in 2013 or later described in chronological order.

The instruction of the Prime Minister at the Headquarters for Japan's Economic Revitalization based on the discussion at the first meeting of the Industrial Competitiveness Council (Jan 25, 2013)

The Prime Minister's instruction to the Minister of Ministry of Economy, Trade and Industry (METI) is to review from scratch "Innovative Strategy for Energy and the Environment" decided by the former administration and establish a robust and responsible energy policy from various perspectives, including stable supply of energy and reduction of energy cost.

Monju Research Plan (September 2013) [8]

In September 2013, the Ministry of Education, Culture, Sports, Science and Technology (MEXT) adopted Monju Research Plan, which summarizes expected outcomes by conducting R&D using Monju and how long does it take to obtain such outcomes from the technical point of view (technological priority and level of importance), based on the current status in Japan and abroad. The plan presented the three main pillars of Monju R&D that aims for the following: (i) Compilation of outcomes of FR development, (ii) Reduction of the amount and toxic level of radioactive waste, and (iii) Safety enhancement of FR.

4th Strategic Energy Plan of Japan (April 2014) [4]

In April 2014, the Cabinet approved the 4th Strategic Energy Plan, which indicates Japan's new direction of energy policy for the next 20 years or so. The plan describes below.

- Nuclear power is an important base-load power source as a low carbon and quasi-domestic energy source, contributing to the stability of energy supply-demand structure.
- Dependency on nuclear power generation will be lowered to the extent possible and the volume of electricity to be secured by nuclear power generation will be carefully examined.
- In case that the Nuclear Regulation Authority (NRA) confirms the conformity of nuclear power plants with the new regulatory requirements, the Government will proceed with the restart of the nuclear power plants.
- The Government will steadily promote a nuclear fuel cycle.
- The Government will promote FR R&D through international cooperation with the US and France, etc. and promote the development of technologies for reducing the volume and harmfulness of radioactive waste using FRs and accelerators in order to secure a wide range of options in the future.
- The Government will position Monju as an international research centre for technological development, such as reducing the amount and toxic level of radioactive waste and technologies related to nuclear non-proliferation.

Long term Energy Supply and Demand Outlook (July 2015) [9]

The Agency for Natural Resources and Energy of the Ministry of Economy, Trade and Industry (METI) approved the "Long term Energy Supply and Demand Outlook" (Long term Outlook) in July 2015, based on the Strategic Energy Plan approved by the Cabinet in April 2014. In the Long term Outlook for 2030, Japan is supposed to achieve an improvement in energy self-sufficiency to around 25% and the reduction of energy costs, as well as Greenhouse Gas (GHG) reduction with a target in line with those of Europe and the United States by promoting energy conservation, introducing renewable energy as much as possible and improving efficiency in thermal power generation, etc. Specifically, the government intends to reduce the GHG emissions by 26% from the FY2013 level, by achieving a share of 20% to 22% of nuclear energy and 22% to 24% of renewables in the electricity generation mix.

NRA's recommendation on Monju (November 2015) [10]

The NRA issued the Minister of MEXT a recommendation that states a qualified management body other than Japan Atomic Energy Agency (JAEA) should be identified to operate Monju safely or review of the status and future of Monju should be conducted.

Report of the MEXT's Panel for Discussion on the Status and Future of Monju (May 2016) [11]

MEXT set up the Panel for Discussion on the Status and Future of Monju (Chair: Mr. Akito Arima) to discuss revolving around Monju in responding to the NRA recommendation. The panel held hearings with related parties and conducted on-site inspections with the main aim of extracting those requirements that a prospective operator of Monju should be able to meet after examining and summarizing problems regarding Monju and issued a report.

The new policy for FR development in Japan and policy on the Monju (December 2016) [6], [7]

After the NRA recommendation concerning Monju to the Minister of MEXT issued in November 2015, the Inter-Ministerial Council in September 2016 decided to establish the Council on FR Development (Council) consisting of the Minister of METI (Chair), the Minister of MEXT, JAEA and private entities (electric utilities and core manufacturers) concerned with FR development, which aims at fundamentally reviewing the Monju project including its decommissioning and discussing future approaches of the FR development in Japan. In response to the discussion of the Council, the Inter-Ministerial Council decided in December 2016 the new policy for FR development (refer to Chapter 4) and the policy on the Monju.

The policy on the Monju describes that various technological outcomes and knowledge at Monju have been accumulated and basic technologies for establishing a system of power generating plant have been acquired as a prototype reactor; however, it was decided that Monju should not resume operation but decommissioned as well as take a new role in the future FR development for the following reasons.

— The expected increase of time and cost for the restart of Monju by the adoption of the new regulatory requirements (it will take at least 8 years to resume operation and cost more than 540 billion yen until the end of operation if it is supposed to operate for eight years (five cycles) including performance tests.)
— There was no alternative Monju operator who accommodates the NRA recommendation.
— After the restart of Monju, it will be expected that useful data for the realization of a demonstration reactor, particularly a loop-type demonstration reactor, are obtained. However, knowledge expected to be gained after the restart of Monju can be obtained through such alternative measures as the utilization of domestic test facilities and international cooperation, and R&D in the next demonstration reactor stage.

3. THE STATUS OF FR CYCLE TECHNOLOGY DEVELOPMENT IN JAPAN

The progress of FR cycle technology development in Japan after FR13 (held in March 2013) is as follows.

3.1. R&D aiming at safety enhancement of FRs with the use of international cooperation

In terms of the safety of FRs, it is important to establish safety standards common throughout the world. In order to achieve high safety goals of Gen IV reactors including the Sodium cooled FR (SFR), the Generation IV International Forum (GIF) is engaged in developing Safety Design Criteria (SDC) that embodies the goals and then Safety Design Guidelines (SDG) to deploy the SDC in design [12]. Japan has actively contributed to developing the SDC and SDG as follows. Japan proposed a draft SDC which consolidated international safety requirements for the design of SFRs in light of lessons learned from the 1F accident, and then the draft SDC was examined by the Atomic Energy Society of Japan, followed by the provision to the discussion in the GIF. The SDC has been already discussed in the GIF-SDC task force and then approved by the Policy Group of GIF and is undergoing reviews by regulatory bodies in GIF member countries and international institutions. The development of the SDG for SFRs is following the same procedures.

In light of lessons learned from the 1F accident, a set of new regulatory requirements in consideration of severe accidents for Japanese power reactor facilities in the R&D stage such as the SFR Monju was enacted in July 2013. Since these new regulatory requirements are due to be revised after taking public comments, etc. into consideration before conducting safety inspections, JAEA set up “Monju Safety Peer Review Committee” consisting of FR experts and drew up the report “Safety Requirements Expected to Prototype Fast Breeder Reactor Monju” [13] taking into account the SFR specific safety characteristics. JAEA submitted this report to the NRA in July 2014 and then asked leading international experts to review with the purpose of validating the content. Incorporating the summary of the results of international review, the report was released in September 2015 [14].

Japan has been engaged in the cooperation with France on the Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) project since 2014 while seeking the possibility of FR development utilizing international cooperation. There are many common SFR technologies in the ASTRID although it is a pool type which is different in configuration of the reactor and the primary coolant system from a loop-type reactor like Monju that Japan has developed so far.

Joyo has conducted operations for 71,000 hours in total and irradiation of around 100 test assemblies so far since it reached first criticality in 1977 [15]. However, Joyo has been shut down since June 2007 because part of the fuel handling function was damaged due to the interference between a small rotating plug for refuelling and the Material Testing Rig with Temperature Control (MARICO-2). JAEA completed the replacement of the damaged Upper Core Structure (UCS) and the retrieval of the bent MARICO-2 sub-assembly in 2014. Then it restored the surroundings of the reactor vessel to a normal state by loading a new UCS on the rotating plug in November 2014 [16] and completed re-installation work of the retrieved equipment on the rotating plug in June 2015. JAEA applied for a change of reactor installation license for restart in around 2020 under the new regulatory requirements at the end of March 2017 and is currently examining safety enhancement measures. In addition, it will discuss the operation plan for Joyo including the utilization of international cooperation to meet requirements in Japan and abroad such as the U.S. and France.

3.2. R&D for reducing volume and harmfulness of high-level radioactive waste by the use of FR cycle

In order to realize the recycling with FRs using Minor Actinides (MAs), it is necessary to separate and recover MAs from spent fuels, fabricate MA bearing fuels using the recovered MAs and irradiate them to verify that MAs can be burned as expected. JAEA has conducted Small Amount of Reused Fuel Test (SmART) project with a small amount of MAs using its R&D facilities at Tokai and Oarai sites and has extracted neptunium together with uranium and plutonium from 4 spent fuel pins irradiated at Joyo so far [17]. It will recover americium and curium from high-level liquid waste (raffinate) and fabricate MA bearing MOX fuel using more than 1 gram of the extracted MAs followed by the irradiation at Joyo for a post-irradiation experiment.

Meanwhile, JAEA has fabricated 301 fuel assemblies for Joyo and 366 fuel assemblies for Monju at its Plutonium Fuel Production Facility (PFPF) [18], where it is preparing for the application for a restart under the new regulatory requirements to supply fuel for Joyo.

4. NEW POLICY FOR FR DEVELOPMENT IN JAPAN

The Council on FR Development (Council) was established based on the decision "Future Approaches to Developing FRs" at the meeting of the Inter-Ministerial Council held on September 21, 2016. Based on the discussion at the Council, the new policy for FR development was decided as a guide for parties involved in the FR development at the meeting of the Inter-Ministerial Council on December 21, 2016 [6]. The policy states that Japan should maintain and develop technical infrastructures at a world class level and develop and commercialize FRs with a high level of safety and economics, and thereby aim to play a leading role towards the realization of international standards. Based on this policy, Japan will integrally carry out the formulation of strategy and establishment of research systems for the realization of FRs in future.

Therefore, it was decided that a practical level strategy working group made up of "International Cooperation" team, "Joyo" team, "Monju" team, "Domestic Facility" team and "Administration" team, which controls the other teams, will be established under the Council to develop "Strategy Roadmap" (tentative name) that identifies developmental work for the coming 10 years in line with basic concepts as follows (Fig. 1).

4.1. Basic approach

As Japan has accumulated reasonable intellectual assets, it is possible to set to focus again on developmental work for the design stage of a demonstration reactor utilizing these assets. Knowledge expected to be gained from the future operation of Monju will instead be obtained by new measures. In the demonstration reactor development with the development goals clarified that it aims to (i) develop individual technologies that can be implemented in the FR plant (element-technology development), (ii) to clearly articulate targets of what kind of plants it aims for and a concept of the plant that meet the targets (specifications including reactor type and scale) and identify the appropriate combination of technologies to realize the plant concept (the determination of the plant concept), (iii) build an integrated plant system that includes peripheral equipment (integrated system design).

For the time being, it will devote resources in the decision of the plant design, make the best use of domestic knowledge and facilities and carry out developmental work by the use of optimal facilities in the international network and the collaboration with appropriate institutions while conducting basic and fundamental research.

5. CONCLUSION

Japan has been carrying out the FR development for the early commercialization of the FR cycle through the stages of the construction and operation of experimental FR Joyo and prototype FBR Monju, and the R&D of a demonstration reactor as a national policy from the initial stage of nuclear energy development. Although it was decided in December 2016 that Monju will not resume operation as a reactor, but set to be decommissioned, Japan intends to firmly maintain the basic policy to promote the nuclear fuel cycle and work on the FR development, and it is determined to develop and materialize the Strategy Roadmap incorporating the active utilization of international cooperation toward the early realization of the FR cycle.

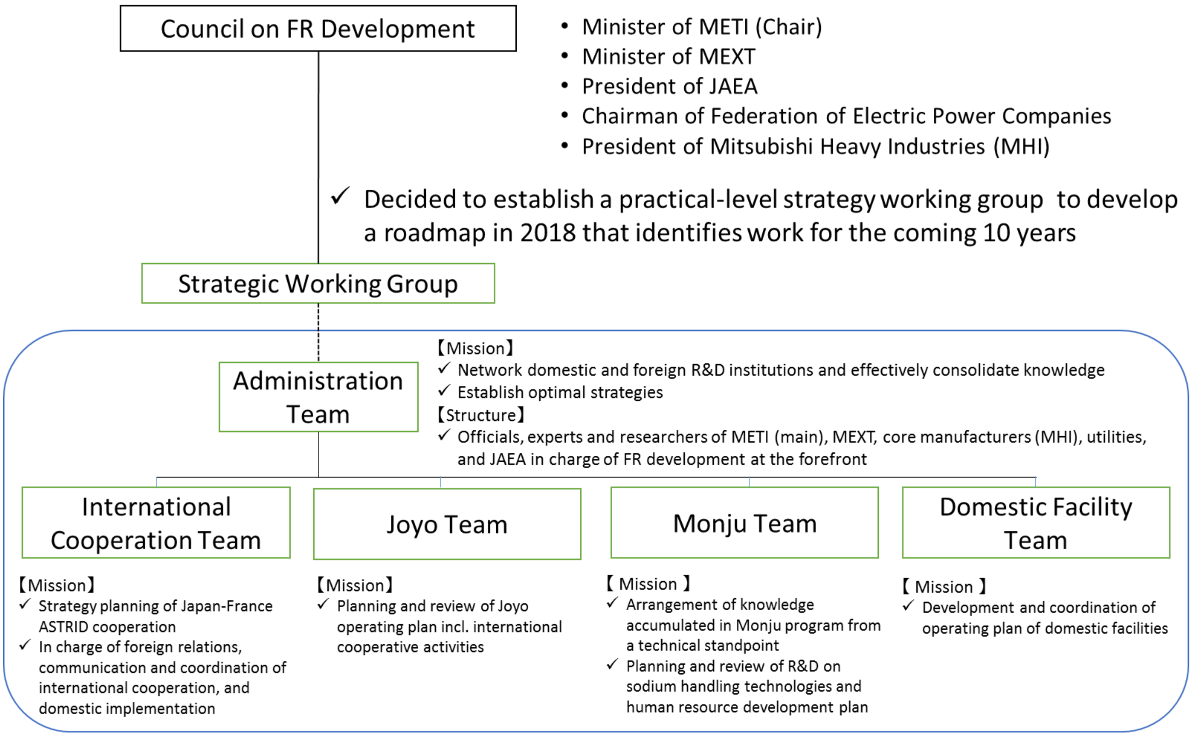

FIG. 1. Organization Structure for Strategy Roadmap (tentative name) on FR Development.

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4.6. REPUBLIC OF KOREA

Status of Sodium Cooled Fast Reactor Development Programme in Korea

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Abstract. The Korea Atomic Energy Commission (KAEC) authorized the R&D action plan for the development of the advanced Sodium cooled Fast Reactor (SFR) and the pyro-processing technologies to provide a consistent direction to long term R&D activities in December 2008. This long term advanced SFR R&D plan was revised by the KAEPC in November 2011 in order to refine the plan and to consider the available budget for SFR. The revised milestones include the specific design of a prototype SFR by 2017, specific design approval by 2020, and construction of a prototype Gen IV SFR (PGSFR) by 2028. The prototype SFR programme includes the overall system engineering for SFR system design and optimization, integral V&V tests, and major components development. Based upon the experiences gained during the development of the conceptual designs for KALIMER, the conceptual design of PGSFR has been carried out in 2012 and has been performing a preliminary design since 2013. The first phase of the development of PGSFR was completed at the end of February 2016 and now going towards the second design phase. All the design concepts of systems, structures and components (SSCs) have been determined and incorporated into the Preliminary Safety Information Document (PSID), which includes basic design requirements, system and component descriptions, and the results of safety analysis for the representative accident scenarios. The PSID will be a base material for a pre-review of the PGSFR safety. The target of the second phase of PGSFR design is to prepare a Specific Design Safety Analysis Report (SDSAR) by the end of 2017. The SDSAR is equivalent to the conventional preliminary safety analysis report (PSAR) but without the specific site information of the plant. To support the design, various R&D activities are being performed in parallel with design activities, including V&Vs of design codes and system performance tests.

Key Words: Sodium cooled fast reactor, Metal fuel, Gen IV, Pool type

1. INTRODUCTION

Light Water Reactors (LWRs) have been operated and played a significant role in a stable electricity supply and economic growth of Republic of Korea since 1978. There are twenty-one LWRs and four CANDU type reactors currently in operation, three PWRs under construction and four additional PWRs planned by 2029 based on the 7th National Electricity Demand and Supply Plan. The construction of nuclear power plant also supports the Paris Agreement on new climate change agreed at COP21 in 2015, which targets nuclear share of 29% by 2035.

One of the serious obstacle in constructing LWRs is a problem on Spent Nuclear Fuel (SNF) management because of high radio-toxicity and long half-life of SNF. Annually about 760 tons of SNF are discharged from PWRs and total stored SNF amounted to 14,468 ton as of December 2015. In this context, the Public Engagement Commission on Spent Nuclear Fuel (PECOS) made 10 recommendations on future spent nuclear fuel management policy in 2015. One of key recommendation among them is to establish an R&D plan for volume and toxicity reduction of SNF. The SNF problem has been a common concern among the countries having utilized nuclear energy for a long time or having a plan to extend the utilization of nuclear energy. The SFR has been widely recognized as a technical alternative to effectively manage the SNF owing to its transmutation capability of long-lived radio-toxic nuclides included in the SNF. It can be accomplished by using abundant high energy excessive neutrons in the core.

For this reason, the SFR development plan is always accompanied with the policy for the extension of nuclear energy in many countries.

The Long term Development Plan for the Future Nuclear Energy Systems was authorized by the Korean Atomic Energy Commission (KAEC) in 2008 and updated by the first KAEPC in 2011. This includes a construction of a prototype SFR by 2028, with the preparation of preliminary safety (PSID) by 2015 [1], issues of specific design safety analysis report by 2017 and its approval by 2020.

In July 2016, the sixth KAEPC has approved the Basic Plan for High-level Radioactive Waste Management, which includes the security of underground research laboratory, interim storage and final disposal sites of high-level waste, enactment of a special law on the procedure for high-level waste management. The KAEPC has also approved the Demonstration Strategy of the Future Nuclear Energy System Development. The strategy provides basic directions for the demonstration of the pyro-processing and sodium cooled fast reactor technologies.

The national project to develop the Prototype Gen IV Sodium cooled Fast Reactor (PGSFR) was initiated to achieve the national mission in 2012. For this, Sodium Cooled Fast Reactor Development Agency (SFRA) dedicated to the PGSFR development was established in the mid of 2012. R&D works of the PGSFR project are mainly carried out by KAERI, KEPCO E&C and Doosan Heavy Industry. KAERI is in charge of the design and the validation of Nuclear Steam Supply System (NSSS) and fuel development, and KEPCO E&C is responsible for the balance of plant system design. Doosan Heavy Industry involves in the evaluation of a mechanical design and fabrication of major components. KAERI is closely working with Argonne National Laboratory (ANL) under an agreement on the joint development programme approved as a Work For Others (WFO) contract. ANL supports KAERI with their experiences in SFR development and is jointly working on the developments of codes for fuel rod performance analysis and severe accident analysis. The collaborative activities through a Generation IV International Forum (GIF) and IAEA Coordinated Research Projects (CRP) also support the R&D activities for the PGSFR development.

2. STATUS OF PGSFR DEVELOPMENT PROJECT

2.1. Design Status of PGSFR

The main goal of the PGSFR development is to demonstrate transmutation capability of Transuranic (TRU) nuclides which are the major long-lived toxic elements included in the LWR spent nuclear fuel. The high level of safety and the efficient electricity generation are also one of the requirements of the PGSFR [2].

The initial core of the PGSFR is loaded with low enriched uranium metal fuel (U-10% Zr) for a reactor performance demonstration and as a driver fuel for TRU fuel irradiation test as shown in Table 1. Several Lead Test Rods (LTRs) and Lead Test Assemblies (LTAs) containing TRU fuel recycled from LWR (LWR-TRU) will be loaded and qualified during this period. After qualification of LWR-TRU fuel, the U-TRU-Zr fuel will be loaded into the core as a batch. Until this stage, the back-end fuel cycle will be kept as once through without self-recycling. The in-reactor performance of the self-recycled TRU fuel (MTRU) will also be demonstrated during the LTRU core operation. Then finally, the fully closed fuel cycle with the self-recycling will be demonstrated in the MTRU core.

TABLE 1. EVOLUTION OF PGSFR CORE

U core	LTRU core	MTRU core
U-10%Zr fuel	U-TRU-Zr fuel	U-TRU-Zr fuel
Open fuel cycle	LWR-TRU equilibrium	TRU core equilibrium with self-recycling + LWR recycling (MTRU fuel equilibrium)
LWR-TRU (LTRU) fuel demonstration	Open fuel cycle	
LTR and LTA test zone installation	LWR-TRU and self TRU mixed (MTRU) fuel demonstration	

Figure 1 shows the key design features and schematic diagram of the PGSFR [3]. The overall design features can be summarized as metal fuelled, pool type sodium cooled fast reactor with active and passive decay heat removal and shutdown systems.

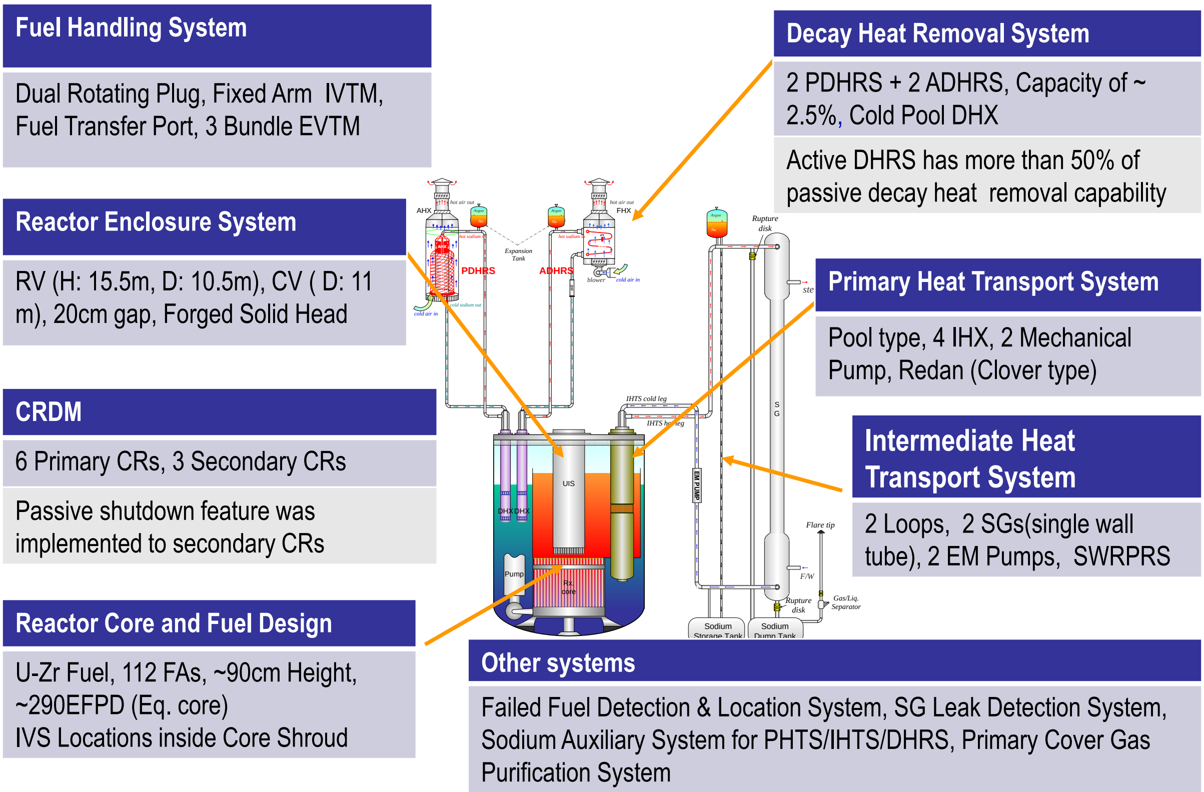

FIG. 1. Key design features of PGSFR.

A total 217 fuel rods are arranged into the fuel sub-assembly with hexagonal configuration. Total 112 fuel sub-assemblies are loaded into the core in the hexagonal configuration. The active core height is about 90 cm. The cycle length of uranium equilibrium core is about 290 Effective Full Power Days (EFPDs). There is neither an axial nor radial blanket to prevent additional TRU production in the blanket region. The active core is directly faced into the steel reflector.

The Primary Heat Transfer System (PHTS) of the PGSFR is a pool type. All the structures and components of PHTS, four Intermediate Heat Exchangers (IHXs) and two mechanical pumps are submerged into a large sodium pool confined by double vessels; reactor vessel and containment vessel.

The Intermediate Heat Transfer System (IHTS) consists of two loops with two steam generators. The annular linear induction pump is used in the IHTS and IHTS is connected into the Sodium-Water Reaction Pressure Relief system (SWRPRS) to prevent over-pressure of IHTS loop when sodium-water reaction occurs in the steam generator. The IHTS pipe inside containment is double-walled. The gap between inner and outer pipes is filled with inert gas and continuously monitored by diverse leak detectors as well as the gap between reactor and containment vessels.

The decay heat is rejected to the atmosphere by Decay Heat Removal system (DHRS). DHRS consists of four trains: two Active Decay Heat Removal systems (ADHRs) and two Passive Decay Heat Removal systems (PDHRs). The sodium to air heat exchangers are a finned tube type for the ADHRSs and a helical coil type for the PDHRSs to satisfy a diversity and redundancy requirements for the safety system. The active functions of ADHRSs are provided by an air blower. The active circuit has also passive function by the natural circulation of sodium and air and its passive heat removal capacity of the active circuit is more than 50% of designed heat removal capacity even when air blowers are not operable.

The PGSFR has the independent and diversified safety shutdown systems consist of six primary Control Rods (CRs) and three secondary shutdown rods. A passive shutdown mechanism is implemented into the secondary shutdown rods for additional shutdown capability beyond design basis accidents. The major design parameters of the PGSFR are listed in Table 2.

TABLE 2. DESIGN PARAMETERS OF PGSFR

Parameters	Value
Core power [MW(th)/MW(e)]	392.2/150
Coolant temperatures (inlet/outlet) [℃]	390/545
Total core flow rate [kg/sec]	1,990
Fuel type (initial and transition)	U-10%Zr
Cycle length [EFPD]	290
Fuel cladding material	FC92(FMS)
Number of batch (inner/outer)	4/5
Active core height [cm]	90
Pitch to diameter ratio (P/D)	1.14
Total heavy metal inventory [ton]	7.33
Discharge burnup (avg./peak) [MW·d/kg)]	66.1/104.7
Average/peak linear power [W/cm]	159.7/323.7
Sodium void reactivity at EOEC [pcm]	-900

2.2. Safety Analysis Results of PGSFR

The safety analysis for the first phase design has been carried out for the representative bounding accident scenarios [4]. The event classification and corresponding safety design acceptance criteria have been established in terms of Cumulative Damage Fraction (CDF) and temperatures. All the results of the safety evaluation satisfy the acceptance criteria with a sufficient margin. The list of the representative events and the acceptance criteria are given in Table 3.

TABLE 3. EVENT SCENARIOS INCLUDED IN PSID AND ACCEPTANCE CRITERIA

Classification	Events	Remarks
Reactivity Insertion	- Maximum velocity withdrawal of single control rod (DBA1) - Reactivity insertion and pump trip by SSE	- Acceptance criteria - AOO: $CDF_{\Sigma AOO} < 0.05$ - DBA1: $CDF_{event} < 0.05$ - DBA2: Fuel T < Solidus T, Cladding T < 1075ºC, No bulk sodium boiling - DEC: Fuel T < Solidus T, No bulk sodium boiling
Undercooling	- Loss of Flow (LOF-AOO) - Loss of Heat Sink (LOHS-AOO) - One Pump Seizure (OPS-DBA1) - PHTS Pipe Break (PB-DBA2) - SBO (SBO-DBA2) - 5 DEGs of SG tubes (DBA2)	
Increase or decrease of PHTS Inventory	- Reactor vessel leak (DBA2)	
DEC	- ATWS (UTOP, ULOF, ULOHS)	
HCDA	- Mechanical energy release evaluation	- Mechanical energy release during whole core melting (100$/s) ~ 16.1 MJ
PSA	- Level-1 PSA	- CDF by internal events ~ 1.2×10^{-9}/Rx-yr
Source Term	- In-vessel source term evaluation	- Non-mechanistic source term but consider fission product retention in sodium

3. R&D ACTIVITIES ON PGSFR TECHNOLOGY V&V

The Validation and Verification (V&V) activities to demonstrate the system performance and safety of PGSFR are in progress in parallel with the design efforts for the PGSFR [5]. The major activities are as follows:

- Reactor mock-up physics test;
- Performance test for the DHRS heat exchangers (STELLA-1);
- Sodium thermal hydraulics integral effect test for PGSFR (STELLA-2);
- Hydraulic performance test of the PHTS pump;
- Performance test for a Finned-Tube Sodium to Air Heat Exchanger (FHX);
- Intermediate Heat Exchanger (IHX) flow characteristics test;
- Reactor flow distribution test;
- Core thermal-hydraulic characteristics test;
- Dynamic characteristics test of the upper internal structures;
- Performance test of the control rod drive mechanism;
- Drop test of the control rod assembly;
- In-service inspection tests of the waveguide sensor for reactor internals, EMAT for the reactor vessel, and the combined sensor for steam generator tubes;
- Irradiation programs on the advanced cladding and fuel materials;
- Fuel assembly mechanical and hydraulic tests.

The supports of V&V activities are essential to demonstrate the PGSFR safety characteristics. The main items of the performance demonstration and computer codes V&V tests were introduced. The current status of each activity was explained with emphasis on the significant test results. A large-scale Sodium Thermal-Hydraulic Test Programme (STELLA) including the STELLA-1 for separate effect tests and the STELLA-2 for integral effect tests are planned. The STELLA-1 produced satisfactory results of heat transfer characteristics of the decay heat exchanger (DHX) and natural draft sodium to Air Heat Exchanger (AHX) in the DHRS and they were used to validate both the heat exchanger design codes and safety analysis code. The STELLA-2 is under construction and it is expected to be the world's unique integral effect test facility for SFR including all the heat transport systems of PHTS, IHTS, and DHRS. The scale of the STELLA-2 system is of 1/5 in length and of 1/125 in volume. The detailed design of STELLA-2 facility has been completed in 2016 and the procurements of several major components will be initiated during 2017.

The similarity test of the model mechanical pump which was performed in STELLA-1 under sodium condition showed a good agreement with the test results in water. Based upon this result, the performance test of a hydraulic model pump reflecting an impeller and a diffuser with a real-sized design data of PGSFR has been conducted in water to generate the test database which will be used for the safety analysis.

The SELFA facility for Forced Draft Sodium to Air Heat Exchanger (FHX) performance test has been constructed and the test operation is ongoing. The Intermediate Heat Exchanger (IHX) performance test is being prepared by utilizing existing STELLA-1 loop and test section of DHX. The test conditions of flow rate and operating temperature are refined by replacing sodium pump and controller in STELLA-1. The modifications of the loop were already completed and the IHX performance test will be completed within 2017. The detailed design of the test facility for reactor pool flow distribution, called as PRESCO is underway which aims the completion by the first half of 2017.

The performance tests for both primary CRs and secondary shutdown assemblies have been completed. These include performance tests of the control rod driving mechanism such as an electromagnet, an abnormal withdrawal prevention part, a gripper, driving motors, and the verification test for the passive shutdown mechanism. The drop tests of the CRA under scram conditions were performed to be compared with those from drop analyses. The drop analysis methodology was verified with the test results and the optimal design is currently underway.

Three types of inspection sensors are currently under development for the safe operation of PGSFR and their basic performance tests have been conducted; the waveguide sensor for reactor internals, EMAT for the reactor vessel and the combined inspection sensor for steam generator tubes. Although the feasibilities of the developed inspection sensors are successfully demonstrated, more efforts should be made to improve the performance of the application to an actual inspection.

For U-Zr fuel, fuel design for PGSFR, and fabrication of all the fuel components and fuel assembly were performed [6]. Verification tests of U-Zr fuel are underway. The fuel cladding of ferritic-martensitic steel, FC92, which has a higher mechanical strength at high temperature than conventional HT-9 cladding was developed, fabricated and is being irradiated in the fast experimental reactor. Barrier such as Cr electroplating on inner cladding surface to prevent an interaction between the metal fuel and cladding during irradiation was fabricated and tested in the reactor showing satisfactory performance. As a first milestone, the performance of U-Zr fuel will be verified and technical feasibility will be demonstrated by 2020.

Cladding tubes of FC92 and HT9 are subjected to irradiation tests in an experimental fast reactor, BOR-60. It is essential not only to demonstrate cladding performance under fast neutron environment but also to measure in-pile characteristics of cladding for fuel design. Irradiation creep and swelling tests are of utmost importance to obtain in-pile creep model of FC92 for which out-of-pile creep data are used as a supplement. Two irradiation rigs were used; (Material Test Rig) MTR-1 and MTR-2 for which nominal irradiation temperatures are 600℃ and 650℃, respectively. Peak irradiation doses at the end of 2019 are expected to reach 45 and 75 dpa for MTR-1 and MTR-2, respectively. In March 2015, MTR-1 and MTR-2 tests were initiated after the completion of the verification test. As of May 2016, peak irradiation dose reached 12.6±1.5dpa in MTR-1 and 22.9±0.7dpa in MTR-2. The first interim inspection was done for MTR-1 and MTR-2.

PGSFR fuel verification is being made through in-pile and out-of-pile tests. Irradiation tests of fuel rods and fuel components are underway in both thermal and fast research reactors. Fuel behaviour depending upon temperature and fission except fast neutron flux can be evaluated by irradiation test in thermal reactor such as HANARO (KAERI) and ATR (INL). The fuel irradiation rig was fabricated, and the irradiation has been begun at an instrumented irradiation position of BOR-60 in July 2016. The in-reactor behaviour of U-Zr fuel rod for PGSFR initial core is scheduled to be mostly confirmed around 2020. The present fuel tests at fast neutron environment will be extended beyond 2020 to reach the target burnup.

4. SUMMARY

The primary goal of PGSFR is the demonstration of reduction of radioactive waste from spent nuclear fuel by transmuting highly radio-toxic and long-lived elements. The successful construction and demonstration of PGSFR will bring Korean nuclear industry a step closer to guarantee the sustainable operation of NPPs nationwide and finally most importantly will solve the issue of spent nuclear fuel management of Republic of Korea.

Based on the experiences gained through the development of past KALIMER designs, KAERI is developing PGSFR design that can better meet the Gen IV technology goals and the technologies necessary for its demonstration. Several advanced design concepts were developed to improve the economics, safety, reliability, and metal fuel performance of SFR in the areas of reactor core, fuel and materials, reactor systems and the balance of plant.

The safety design of PGSFR emphasizes accident prevention by enhancing inherent safety characteristics and passive safety features using natural phenomena. To support the development of PGSFR design and technologies, R&D activities are being performed for various topics including the validation of neutronics analysis codes, safety demonstration of DHRS in conjunction with primary systems, sodium technology development, and metal fuel qualification.

ACKNOWLEDGEMENT

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4.7. RUSSIAN FEDERATION

Russia's Research and Pilot Fast Reactors as the Basis for the Development of Commercial Reactor Technologies

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Abstract: Since the beginning of the 1990s, the Russian Federation has conducted R&D and design activities to develop lead bismuth and lead cooled fast reactors with inherent safety. The development activities related to the fast sodium reactors have been continued and, so far put into operation the BN-800 commercial power reactor with a hybrid core operating oxide and MOX fuel; the BN-1200 commercial fast sodium reactor project was developed as well. The paper discusses the experience in utilizing the research infrastructure as well as its prospects to develop fast reactor technologies.

Key words: Fast Research Reactor, Sodium Cooled Fast Reactors, BOR-60, MBIR

1. INTRODUCTION

As early as the exploration of nuclear energy based on the fission of heavy atoms, it became obvious that nuclear breeding was likely to become one of the conditions both for nuclear energy wide-scale application and future development of the basic tool for civilization's energy safety. The leading countries of the Nuclear Club have established research centres to meet the challenge. They are the Argonne National Laboratory in Idaho (USA), Institute of Physics and Power Engineering in Obninsk and Research Institute of Atomic Reactors in Dimitrovgrad (Russian Federation), and Research Center in Cadarache (France). These centres are where the key research facilities have been constructed, and they still remain operational, enabling tests to be performed and experimental data to be generated necessary to create and develop nuclear technologies.

2. RESEARCH INFRASTRUCTURE

The USSR started developing Fast Reactors (FRs) in the mid-1950s. For a very short period of time, several research reactors were designed and constructed, such as the BR-1 (1955), BR-2 (1956), and BR-5 (1958). In the 1960s, there were commissioned critical facility BFS-1 to simulate the FR neutronic characteristics and BFS-2, one of the world's largest critical facilities. In 1969, sodium cooled fast reactor BOR-60 with a steam turbine to produce electricity was put into operation intended for tests and try out of fuel, structural materials, coolant, systems and equipment.

For the next eleven years, BN-350 (1972–1973) and BN-600 (1980) were commissioned. As a result, the USSR took a leading position in the FR development and operation [2]. From more than 400 reactor-years of the FR operation, the Russian Federation operates its reactors (Table 1), and BOR-60 is the leader in terms of reactor lifetime being under accident-free operation for 48 years [3].

TABLE 1. RESEARCH AND PILOT FAST REACTORS

Reactor	Country	Reactor type	Thermal (electrical) capacity, MW	Start up (year)	Shut down (year)	In operation (years)
EBR-I	USA	Research	1.4 (0.2)	1951	1963	12
BR-5	Russian Federation	Research	5 (0)	1958	1971	12
DFR	GB	Pilot	60 (15)	1959	1977	18
EBR-II	USA	Pilot	62.5 (20)	1963	1991	28
EFFBR	USA	Pilot	200 (61)	1963	1975	12
Rapsodie	France	Research	40 (0)	1967	1983	16
SEFOR	USA	Research	20 (0)	1969	1972	3
BOR-60	Russian Federation	Research	60 (12)	1969	Under operation	48
BR-10	Russian Federation	Research	8 (0)	1973	2002	29
BN-350	USSR (Kazakhstan)	Pilot	350 (130)	1972	1999	27
Phenix	France	Pilot	563 (250)	1973	2009	36
PFR	GB	Pilot	650 (250)	1974	1994	20
KNK-II	Germany	Research	58 (20)	1972	1991	19
JOYO	Japan	Research	50-75/100 (0)	1977	2007	30
FFTF	USA	Research	400 (0)	1980	1992	12
SuperPhenix	France	Pilot	3000 (1240)	1985	1998	13
FBTR	India	Research	40 (13)	1985	Under operation	32
MONJU	Japan	Pilot	714 (280)	1994	1995	1
CEFR	China	Pilot	65 (25)	2014	Under pilot operation	3
PFBR	India	Pilot	1250 (500)	Start-up and adjustment		
MBIR	Russian Federation	Research	150 (40)	Under construction		

At the end of the 20th century, accidents happened in 1979 at the Three Mile Island NPP (USA) and in 1986 at the Chernobyl NPP (USSR) caused stagnation in nuclear engineering, resulted in a drastic reduction of research programs and shutdown and then decommissioning of a significant number of research reactors. As of today, about 30 research reactors worldwide have long term research programs while others are used from time to time or are under a long term shutdown. About 15 research reactors, mainly Materials Test Reactors (MTR), are used to test materials for nuclear engineering. At that, only two fast reactors, Russia’s BOR-60 and India’s FBTR, are used for large-scale testing of fuel and structural materials.

After the severe accident at the Chernobyl’s 4th unit in 1986, the Soviet Union’s intensive nuclear energy development programme was suspended, and the next two decades were devoted to the in-depth research of reactor safety issues, development of new reactor materials and design concepts, as well as enhancement of the promising reactors performance.

The Russian Federation has successfully completed R&D for the recently commissioned BN-800 power reactor with a hybrid core operating uranium oxide and MOX fuel; the BN-1200 commercial sodium cooled FR project is almost finalized, and ongoing R&D has been launched in lead bismuth and lead cooled FR with inherent safety.

As for the research reactors, the BOR-60 lifetime has been extended till 2020 to continue the in-pile testing of the structural and fuel materials; a design of a new fast research reactor MBIR was developed and its construction has been started to further develop and experimentally support the wide-scale programme for commercial power reactors of the next generation.

As compared to other countries, the Russian Federation has extensive reactor capabilities to carry out tests and master technological solutions to introduce new commercial sodium cooled fast reactors. Table 2 presents the key characteristics of world's reactors under operation.

TABLE 2. FAST REACTORS UNDER OPERATION

Reactor	BOR-60	BN-600	BN-800	FBTR	CEFR
Country	Russian Federation			India	China
Start up	1969	1980	2015	1985	2010
Capacity, MW (electrical / thermal)	12/60	600/1470	880/2150	13/40	23/65
Neutron flux density, 10^{15} $cm^{-2}s^{-1}$:					
– max	3.0÷3.6	6.5	6.0	3.4	3.2
– average	2.2÷2.5	4.3	4.0	2.5	2.1
Fuel	UO_2, UO_2-PuO_2	UO_2	UO_2, UO_2-PuO_2	UC-PuC	UO_2 (UO_2-PuO_2)

3. RESEARCH AND ENGINEERING MISSIONS IN DESIGNING FAST REACTORS

As regards water cooled MTR, they did not provide the required flux density and neutron hardness to test materials and structural components of fast reactor cores. Therefore, to study the behaviour of materials under high dose irradiation, to develop high radiation resistant materials and to select the most suitable coolant regarding its thermo-physical parameters and physical-chemical properties, Russia and other countries are developing nuclear engineering, designed and commissioned research and pilot fast reactors.

The neutron flux density in a fast reactor core is an order of magnitude higher than in thermal reactors. Therefore, neutrons with higher energy in the fast reactor core cause more structural changes and damages in structural materials and fuel. It should be mentioned that a fast reactor is able to simulate all the operational conditions and factors affecting core components and materials (neutron flux, temperature, corrosion and mechanical impact of coolant, cyclic loads, etc.) in order to estimate their behaviour under such conditions.

Traditionally, the following activities are carried out at fast research reactors BR-5, BR-10 and BOR-60:

- Irradiation of promising fuels, absorbers and structural materials, and justification of their performance;
- Research in the improvement of the existing designs of fuel elements, control rods, fuel assemblies, etc.;
- Simulation of steady state and transient operational conditions to test components of nuclear reactors;
- Tests of new gages intended to monitor the conditions of reactor, fuel assemblies and coolant;
- Tests of new reactor process systems and equipment.

Almost a half century of experiments and basic research helped most of the key research challenges in sodium cooled FR. Specific studies are currently carried out to reduce errors and detail the existing data.

As of today, the major issues related to the nuclear physics, reactor physics, thermal physics, hydrodynamics of sodium coolant, sodium systems and reactor equipment, have been mostly resolved. The basic mechanisms of physical processes occurring in fast reactors have been almost precisely identified to ensure the reliable design and safe operation. The scope of research in the above areas tends to decrease.

However, there are under researched issues, such as an intensive effect of fast neutrons on materials (radiation physics), physical and chemical processes in burnable nuclear fuel with accumulation of a large amount of fission fragments, and the fuel to cladding boundary. As a result, the burnup value, the main indicator of the fuel cycle efficiency, is far low even by considering the data generated over the decades of operation.

4. INITIAL R&D IN SODIUM COOLED FAST REACTORS

Thermal Physics and Hydrodynamics of the Coolant. The coolant requirements were primarily determined by its thermo-physical properties. The sodium heating in fast reactors is much higher than water heating in thermal reactors, and heat removed from the core is over four times more. As of today, sodium is the most extensively studied coolant for fast reactors having high thermal conductivity and good volumetric heat capacity as well as a rather high heat transfer coefficient that ensures a small temperature drop at the cladding-coolant boundary. In addition, sodium nuclear physical properties, such as a small capture cross-section and low moderating efficiency, are also very attractive. As for hydrodynamics of molten sodium, it is well-examined, too, in particular at sodium cooled stands under thermal hydrodynamic testing of FAs with the actual geometry considering process variation of parameters and strain during operation. Today, the published sources have all the necessary data for thermal physical and hydrodynamic calculations of sodium cooled fast reactors.

Radiation Damage Physics. Structural Materials. Irradiation of structural materials in an intense fast neutron flux results in substantial changes in the mechanical properties and in strain appearance. When investigating the issue, a wide range of factors have been identified affecting the significance of radiation effects; initial material state and structure that depend on thermal and mechanical treatment, impurities, irradiation conditions (temperature and neutron flux), mechanical stress, etc. Such a variety of parameters are often difficult to control cause huge differences in the data, and sometimes even contradicting results.

The key centres of reactor materials science and radiation physics in the USSR were IPPE and RIAR with fast reactors and well equipped hot cells to focus on high-temperature strength, embrittlement and crack resistance of austenitic stainless steels under irradiation, temperature dependence of the vacancy swelling and irradiation creep effects at the initial stage, lifetime parameters of the fuel elements and FAs. Over the past years, we have travelled a long way from the well-known steel grade Kh18N8 (X18H8) and at present, radiation-resistant ferritic-martensitic steels are the priority for the further development. They keep plasticity and strength even under long term irradiation within a high temperature range.

Fuel. Today, the behaviour of UO_2 and MOX ($(UPu)O_2$) fuel under irradiation is most well-studied. Both fuels behave in a similar way under irradiation. Macro-structural transformations (recrystallization) in fuel under irradiation have been examined well along with fuel swelling and migration of fuel fragments.

Experiments have been performed with other fuels, such as uranium mono-carbide, uranium nitride, plutonium oxide, and metal fuel. In single experiments, the burnup comparable to the one in oxide fuel has been achieved.

Equipment. Different reactor components have been successfully tested at research reactors BR-10 and BOR-60, such as intermediate heat exchangers, mechanical and electromagnetic pumps, cold traps, shut-off valves, piping bellows, etc.

BOR-60 became the platform for lifetime testing of several steam generator types. For instance, steam generator PG-2 was operated under the conditions similar to the commercial reactor ones. This was a large-size model of steam generator PGN-200M of the BN-600 reactor. It had no intermediate steam super-heater module, and the heat transfer tubes were shorter. This steam generator was tested during 1978-1982. As a result, the main PGN-200M engineering solutions were verified, operational parameters were specified under steady-state and transient conditions, and the key operational modes, such as heat-up, start-up and shut down, were tested. For the first time thermal and hydrodynamic steam generation processes in multi-tube straight bundles were investigated using such scale model. Investigations were carried out on distribution and composition of the sediments on the heat exchange surfaces as well as on the state of the heat transfer materials. In addition, a wide range of investigations on reverse steam generators were carried out at BOR-60 and other RIAR’s facilities covering long term operation of two Czechoslovak reverse steam generator models.

Safety. During BR-10 and BOR-60 operation, a variety of safety-related experiments have been performed. Among them is gas supply in the core, sodium boiling, flow rate blockage in the experimental FA with fuel element fracture, inter-circuit leakage in steam generators, etc. Several anomalies like absorber rod destruction, displacement of the rod in a weakened gripping device, emersion of the FA in the core have been detected. A thorough study of different normal and abnormal conditions at BOR-60 has enabled the development and adaption of the methods and means to detect anomalies, such as acoustic imaging under sodium layer, reactivity balance calculation systems, as well as parametric, vibro-acoustic and noise diagnostic systems.

A review of the sodium technology, irradiation parameters and reactor characteristics has enabled the development of methods and means to monitor and improve radiation environment, reactor safety, and sodium handling: three-channel sipping control system (activity of gas and sodium, delayed neutrons); highly efficient compact absorbers for sodium purification from caesium radionuclides; decontamination of equipment after contact with sodium and destruction of non-draining sodium from equipment removed out of service or decommissioned; system of recovering cold oxide traps enabling their operation without being replaced.

Longstanding experience in utilizing Russian critical test facilities, fast research (BR-5, BR-10, BOR-60) and power (BN-350, BN-600) reactors demonstrates the reliability and safety of the applied engineering and technical solutions.

In particular, research performed at BOR-60 has made it possible to enhance the reactor safety and achieve the outstanding fuel burnup, justify the performance of different steam generators, control and safety rods, structural materials and fuels. Table 3 presents a list of materials irradiated in BOR-60.

TABLE 3. MATERIALS IRRADIATED IN BOR-60

Material		Type
Fuel	Ceramics	UO_2, UO_2-PuO_2, $(U,Pu)O_2$, UC, UN, UPuN, UPuCN, PuO_2-MgO, NF+Np, Am
	Metal	U, UPu, UZr, UPuZrNb, PuN-ZrN
	Metal ceramics	PuO_2-U, UO_2-U, UO_2-PuO_2-U, UN-U
	Composite	(UPuZr)C, UO_2-NiCr
Absorbing	Samples	B_4C, Ta, Hf, Dy, Sm, Gd, AlB_6, AlB_{12}, Eu_2O_3, HfH_x, Gd_2O_3, Dy_2O_3-HfO_2
	Control rods	CrB_2, B_4C (^{10}B – 19÷80 %), Eu_2O_3, Eu_2O_3+ZrH_2
Structural	Steels	OX18H9 (OKh18N9), X18H10T (Kh18N10T), EP450, EP823, 03X16H9M2 (03Kh16N9M2), EP912, EI847, EP172, ChS68, BX24 (VKh24), ЭП302Ш (EP302Sh), 09Г2С (09G2S), ARMCO, SS316, ODS-(12,14,18)Cr, T91, T92, S421 (HT9), 15-15Ti, 800H, 14YWT, 800H
	Refractory	V, W, Mo, Nb, WC, SiC/SiC
	Alloys	PE-16, X20H45M4Б (Kh20N45M4B), VZrC, E110, E125, E117, E635, VCrTi, FeCr, ZrNbSnFeO
	Moderators	ГРП-2-125 (GRP-2-125), МП6-6 (MP6-6), ГР-280 (GR-280), ARV, IG-11, PGI, ZrH_x
	Neutron sources	Po-Be, Be, Sb-Be
Coolant	Liquid metal	Na, Pb, Pb-Bi
Electrical	Isolation	Al_2O_3, SiO_2, Si, mica
	Cables	mineral-insulated thermocouple cables, heat-resistant mineral-insulated cables
	Magnets	Alnico
Other	Special ceramics	ГБ-7 (GB-7), ИФ-46 (IF-46), PZT, $LiNbO_3$
	Bio-shield	Concrete

Based on research in structural materials, several steel grades and zirconium alloys have been optimized to reduce their irradiation-induced embrittlement, swelling and growth. The research data have been used for other fast reactors (BN-350, BN-600, CEFR) and thermal reactors (VVER, RBMK). They are also used in designing promising fast reactors, such as BREST, SVBR, BN-1200, Myrrha, TW, etc. New calculation and experimental procedures have been developed and introduced along with irradiation rigs including capsule-type rigs, dismountable materials assemblies and experimental FAs, autonomous instrumented channels with different coolants, etc.

However, despite all successful efforts and achievements, several issues still remain unsolved. Among them are those related to promising fuels, structural materials and coolants, significant increase in the fuel burnup and damage dose, transmutation of actinides, and reactor lifetime extension.

5. DEVELOPMENT OF THE RESEARCH INFRASTRUCTURE FOR PROMISING FAST REACTORS. MBIR REACTOR

At the beginning of the 21st century, promising Gen IV nuclear reactors began to be developed worldwide [4]. These new reactors are mainly designed as commercial prototypes that use innovative engineering solutions and materials, thus enabling absolute operation safety, including prevention of core melting, natural coolant circulation, risk minimization and severe accident consequences mitigation with the help of technical means.

India makes intense efforts in a commercial-scale assimilating of sodium cooled fast reactors. The 500 MW(e) PFBR reactor will achieve the criticality in the coming years. After that, several similar units will be constructed as well. China also has a programme for sodium cooled fast reactor construction to have up to 16% of electricity produced at NPPs by 2050.

France carries out research on a comparative analysis of fast reactor concepts, including the arrangement, FA design, core characteristics, steam generator, refuelling system and safety. The research is planned to be finished by 2020 with its further implementation in the 600MW(e) ASTRID reactor to be constructed after 2020.

Republic of Korea develops a 600 MW(e) demo sodium cooled fast reactor SFR (KALIMER) to be operated on metal fuel with minor-actinides additives. The project will be implemented by 2028. A large-scale production of mixed metal fuel will start in 2025. The pyrochemical reprocessing of spent fuel has been started. A lab-scale facility is to be commissioned in 2016 and the prototype one in 2025.

USA focuses on a long term fuel cycle program; nowadays activities on a minor-actinides burner reactor are being carried out in cooperation with Republic of Korea and Japan. The research programme is aimed at improving its feasibility and enhancing its safety. A private company TerraPower is developing a 500 MW(e) demo sodium cooled fast reactor to be operated on metal fuel; it is based on a “travelling wave” concept to have high burnup due to fissile isotopes breeding [5].

Belgium proceeds with the development of reactor MYRRHA-ADS with lead bismuth coolant. This reactor is to function both under subcritical conditions operated by an accelerator and under critical ones.

The Russian Federation is known to have a Federal Target Programme “Nuclear Power Technologies of the New Generation for 2010 - 2015 and until 2020” under which a design of a 1200 MW(e) commercial sodium cooled BN-1200 reactor has been developed; 300 MW(e) BREST-OD-300 reactor and lead cooled BN-1200 reactor designs are under development; nitride fuel manufacturing facilities are being constructed along with pilot SNF reprocessing facilities.

The development of safe and competitive nuclear facilities of the new generation will require a large scope of in-pile tests and Post Irradiation Examination (PIE) of new materials and core components in specific experimental rigs and loops equipped with cutting edge control means. Structural materials must provide for the reliable operation of core components up to damage dose of 170 dpa. The reactor safety must be experimentally justified under the transient, cyclic and accidental conditions. Therefore, in the frame of the above Program, the infrastructure projects are implemented on the design and construction of multi purpose fast research reactor

MBIR and Poly-Functional Radiochemical Complex to carry out research in SFN reprocessing and radioactive waste handling technologies.

Research reactor MBIR is being constructed at the JSC “SSC RIAR” site and will become a worthy successor to reactor BOR-60 having much better experimental capabilities. Once MBIR is commissioned, the experimental programs will be transferred into it from BOR-60 [6]. Table 4 gives BOR-60 and MBIR characteristics.

TABLE 4. KEY CHARACTERISTICS OF BOR-60 AND MBIR REACTORS

Parameter	BOR-60	MBIR
Nominal thermal capacity, MW(th)	60	150
Max neutron flux density in the core, $cm^{-2} \cdot s^{-1}$	$3.5 \cdot 10^{15}$	$5.3 \cdot 10^{15}$
Core height, mm	450	550
Number of cells in the core for materials testing assemblies	14	27
Across-flats size of core cells for materials testing assemblies, mm	44	72
Number of instrumented cells in the core	1	3
Number of cells to install beyond-the-vessel loop channels	-	3
Across-flats size of cells to install beyond-the-vessel loop channels, mm	-	128
Fuel in standard FAs	$(UPuO_2)$ MOX	$(UPuO_2)$ MOX
Primary coolant T: – reactor inlet, °C – reactor outlet, °C	310–340 510–540	330 512
Scheduled lifetime	2020	2070
Capacity factor	0.65	0.65
Duration of operation between refuelling, eff. days	Up to 100	Up to 100
Time for outages, including refuelling, days	35–45	35 – 45

MBIR has much greater irradiation capabilities as compared to BOR-60. The maximal neutron flux density in the MBIR core is ~ 1.5 times higher and physical volume for irradiation is three times larger as compared to BOR-60.

The key activities to be done at MBIR are:

— Large-scale in-pile testing of promising structural materials in the temperature range up to 1800ºC in different environments for the next generation nuclear facilities, including fusion ones;
— Large-scale in-pile testing of dummy fuel elements with promising fuels for the next generation nuclear facilities;
— Reactor experiments related to the closed fuel cycle, including minor-actinides burning and reduction of radioactive waste;
— Investigation of behaviour of nuclear fuels and justification of their performance under transient, cyclic and accidental conditions in loop facilities with different coolants;
— Applied and medical-purpose research in horizontal and vertical channels.

Thus, the commissioning of the MBIR reactor will allow performing all foreseeable experiments on the development and justification of innovative nuclear power facilities.

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4.8. EUROPEAN COMMISSION (EC)

European Commission Contributions to the Development of Safe and Sustainable Fast Reactor Systems

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Abstract. In Europe, the contributions to the development of fast reactor systems are based on research programs organized in the frame of the EURATOM contract. The European Sustainable Nuclear Industrial Initiative (ESNII) addresses as one of the three pillars of the Sustainable Nuclear Energy Technology Platform (SNETP) the development and deployment of Gen IV fast neutron reactor technologies, together with supporting research infrastructures, fuel facilities and R&D work. The Joint Programme on Nuclear Materials (JPNM) of the European Energy Research Alliance (EERA) makes essential contributions in the fuels and materials development area. These tools organize the distribution of R&D resources, the activities are organized in multi annual framework programs (at present Horizon 2020). Collaboration projects co-financed by the European Commission department responsible for EU policy on Research, Science and Innovation (DG RTD) and direct research programs are carried out in the directorate for Nuclear Safety and Security of the Joint Research Centre (JRC). The focus is on the reactor itself, the reference being the Sodium cooled Fast Reactor (SFR), alternatives the lead and gas cooled systems (LFR and GFR), on the fuels, the fuel cycles and the materials. Furthermore, JRC is the implementing agent for EURATOM in the Generation IV International Forum (GIF) with contributions to all six systems selected for further R&D, task forces and cross-cut working groups.

1. INTRODUCTION

The European Strategic Energy Technology Plan (SET-Plan) aims to transform the way energy is produced in the EU with the goal of achieving EU leadership in the development of technological solutions capable of delivering 2020 and 2050 energy and climate targets.

A recent update of the SET-Plan [1] claims that nuclear power is set to make an ongoing contribution to the decarbonization of the European energy system and achieving the ultimate goal of reducing Europe’s dependency on fossil fuels.

In this context, the Sustainable Nuclear Energy Technology Platform (SNETP) was officially launched in September 2007 to promote research, development and demonstration of the nuclear fission technologies necessary to achieve the SET-Plan goals. Major EU technology challenges for the next decades were defined in the Steering Group meeting on 14 September 2016 with an agreement on strategic targets and priorities such as a specific target on ‘Innovative Emerging Technologies' including Gen IV systems.

The legal basis for the nuclear activities in Europe is the EURATOM treaty setting up a European Atomic Energy Community. It was signed in Rome in March 1957, has the general objective to contribute to the formation and development of Europe's nuclear industries, to ensure security and high safety standards and to prevent nuclear materials from being diverted to military use.

Today half of the EU Member States have decided nuclear energy to be part of the energy mix, ensuring in their country the security of electricity supply. In this context, these Member States are committed, based on the European Nuclear Safety Directive, to apply the highest standards of safety and security. Together with an adequate waste management and non-proliferation as well as diversification of nuclear fuel supplies help to achieve the objectives of the 2030 climate and energy framework in Europe (Fig. 1). With 27% of electricity produced from nuclear energy and 29% from renewable sources, the EU is currently one of the three major economies that generate more than half of their electricity without producing greenhouse gases.

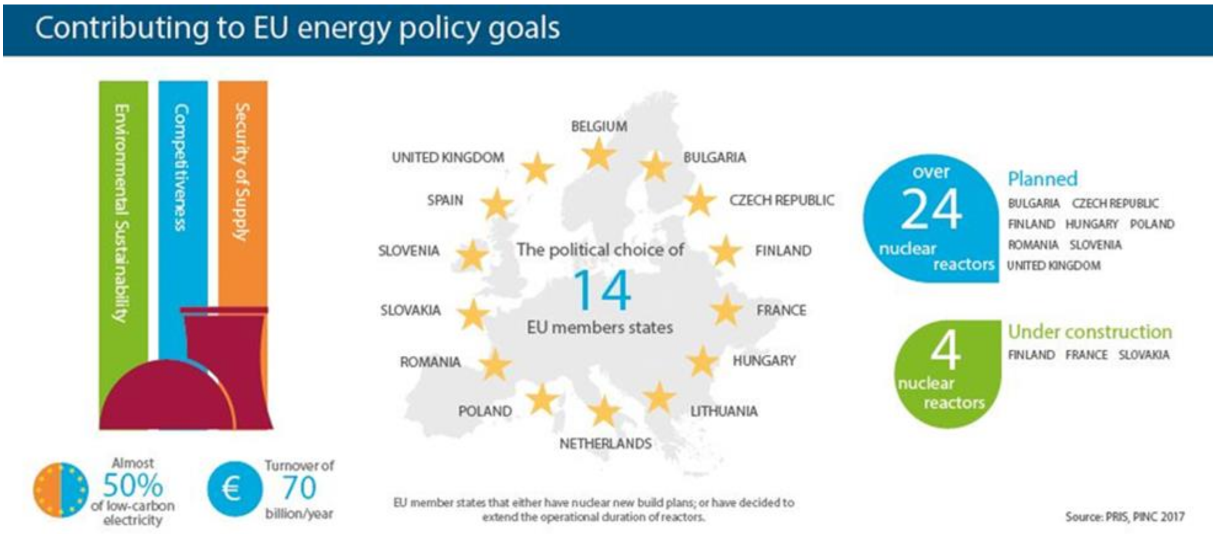

FIG. 1. Nuclear Contribution to Low Carbon Energy Production [2].

The 2016 edition of the Nuclear Illustrative Programme (PINC2016), a communication to be made by the European Commission as foreseen in Article 40 of the EURATOM Treaty, provides a recent overview of investments in the EU for all the steps of the nuclear lifecycle. PINC2016, the first after the Fukushima Dai-ichi accident in March 2011, is supposed to bring clarity on long term (2050) nuclear development in Europe. Specifically, Gen IV is being included in the R&D programs to foster the implementation of advanced reactor systems.

Experts worldwide agree that all low-carbon energy technologies, including nuclear power, are needed to meet the Paris Agreement (COP21) goal set by world leaders in November 2015 to limit the rise of global temperatures to below 2°C [3].

2. EUROPEAN SUSTAINABLE NUCLEAR INDUSTRIAL INITIATIVE (ESNII)

Key elements of Europe's SET-Plan are European Industrial Initiatives (EIIs), industry-led programmes with the aim to boost research and innovation and to accelerate deployment of the technologies. In the nuclear field, ESNII is one of the three pillars of SNE-TP and in this context a task force comprising research organizations and industrial partners. It was launched at the SET-Plan conference in Brussels on 15 November 2010. ESNII addresses the need for demonstration of Gen IV fast neutron reactor technologies, together with the supporting research infrastructures, fuel facilities and R&D work.

In its Strategic Research & Innovation Agenda, ESNII has prioritized the different Gen IV systems and proposed to develop the following projects:

- The sodium cooled fast neutron reactor technology (the ASTRID project) as the reference option, with the construction of a prototype around 2040 in France;
- A first alternative technology is the lead cooled fast reactor (ALFRED) with the construction of an experimental reactor to demonstrate the technology, in another European country willing to host this programme, and supported by a lead bismuth

cooled so called Accelerator Driven System (ADS) an irradiation facility called MYRRHA in Belgium;

— As a second alternative technology, the gas cooled fast reactor (ALLEGRO), also requiring the construction of a technology demonstrator in a European country.

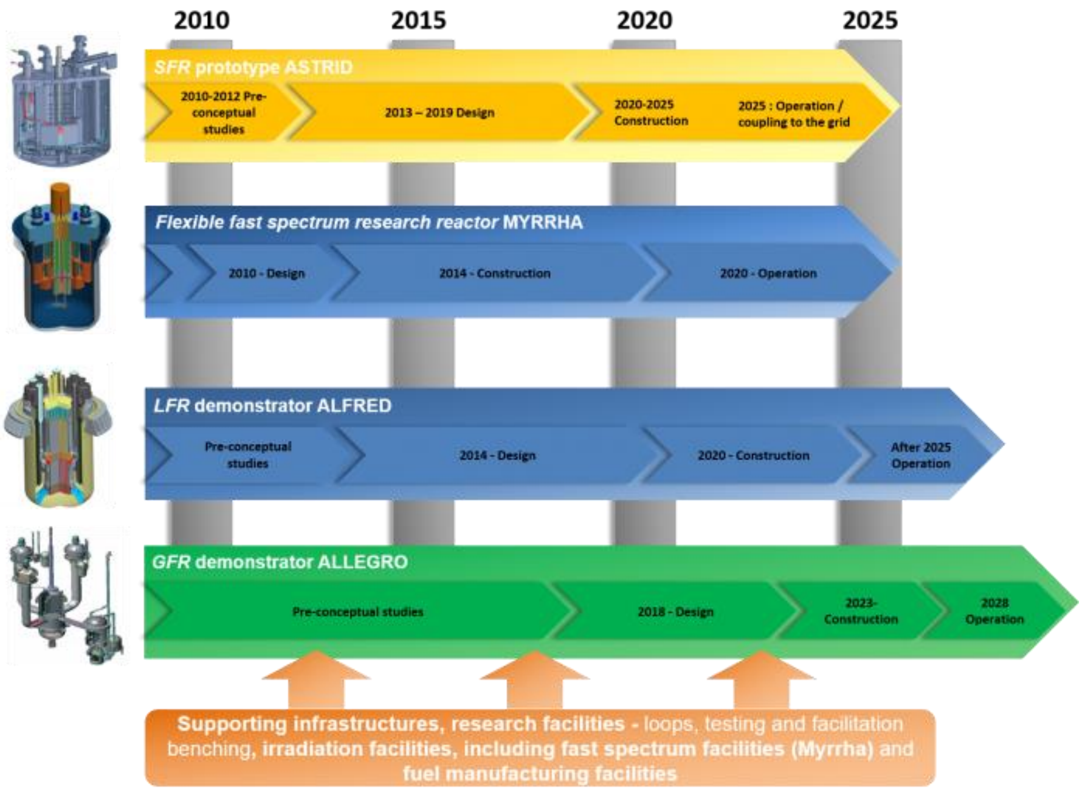

FIG. 2. ESNII roadmap [4].

Within ESNII a roadmap (Fig. 2) has been elaborated including demonstrators, prototypes, main research and testing facilities, in terms of design, planning, budget, legal issues, and intellectual property. Support infrastructures, research and testing facilities including fast spectrum manufacturing will be irradiated and fuel will have to be developed in parallel.

3. THE EUROPEAN ENERGY RESEARCH ALLIANCE JOINT PROGRAMME ON NUCLEAR MATERIALS (EERA- JPNM)

A key pillar of the SET-Plan is EERA (European Energy Research Alliance) which is an alliance of the present 175 leading energy research institutes (research centres and universities) from 27 countries in Europe. Its main objective is to coordinate and accelerate the development and market deployment of new energy technologies. The core of EERA´s activities are the 17 so called Joint Programmes.

In the nuclear field, the EERA Joint Programme for Nuclear Materials (JPNM) was launched in November 2010, with the aim to integrate research activities at European level based on the joint and to identify key priority materials research topics, in support of the development and optimisation of sustainable nuclear energy systems (Fig. 3). The activities are focussed on six main topics:

— Materials for ESNII demonstrators and prototypes;

- Innovative high-temperature resistant steels;
- Refractory materials: ceramic composites, cermets and metal based alloys;
- Physical modelling and modelling oriented experiments for structural materials;
- Synthesis, irradiation and qualification of advanced fuels;
- Physical modelling and separate effect experiments for fuels.

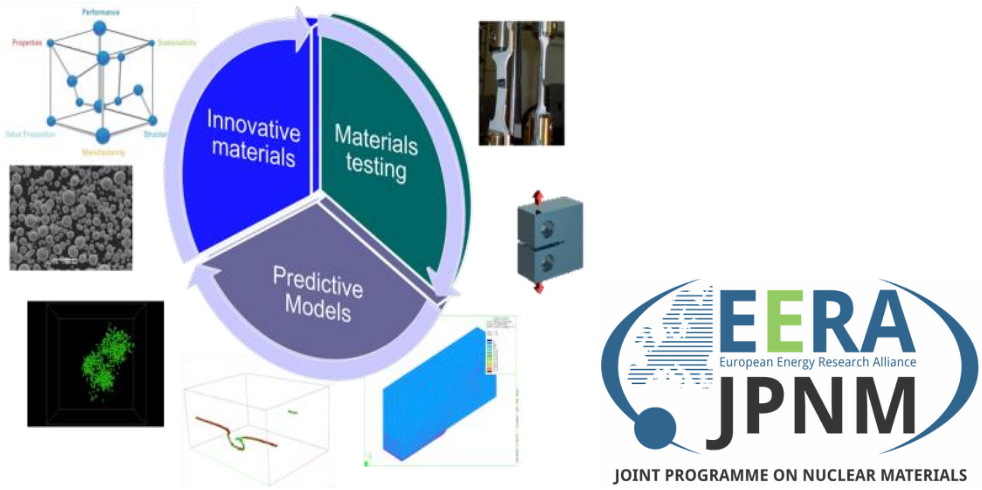

FIG. 3. Research activities in the EERA-JPNM [5].

Especially for Gen IV systems, the operating conditions envisaged are more demanding with respect to the performance of structural and fuel materials compared to presently running commercial reactors. Therefore, the safety and the operation of most of these nuclear system concepts will depend crucially on the capability of the chosen materials to withstand the expected highly demanding operating conditions.

The activities involve structural materials and fuels and rely on basic physical modelling up to material testing, for instance refractory materials, innovative steels and advanced fuels, the final goal being the industrial application.

4. EURATOM AND GIF

In July 2003 EURATOM adhered to the Generation IV International Forum (GIF). Since January 2006 the JRC is entrusted with the coordination of the European Community contributions to GIF, representing all EU Member States except France, which has an own membership. EURATOM is an active member in all 6 systems selected by GIF for further R&D, participating in the steering boards and in a large number of respective research programs.

On 10 November 2016 Commissioner Navracsics has signed for EURATOM the extension of Framework Agreement for additional 10 years.

DG RTD reinforces mutually research efforts of the Member States and the private sector in support of GIF through their so-called indirect action programs. In the 3 framework programs since 2006, 39 related projects with a total EC contribution of about € 143 million were co-funded with the following approximate distribution on:

- Reactor systems: 11 projects - € 35 million;

— Advanced fuel and fuel cycle: 10 projects - € 49 million;
— Advanced materials: 9 projects - € 30 million;
— Codes and data: 9 projects - € 29 million.

Besides this research programs directly connected to the GIF system development other research programs on research infrastructures or on education, training and competence building are of course highly relevant also for the development of GIF systems.

A very important project regarding the development of Gen IV systems in the EU is ESNII+, because it involves reference and alternative options, aiming to prepare a proper ESNII structuration and deployment strategy. An efficient European coordinated research on reactor safety for the next generation of nuclear installations should be ensured and a strong link with the SNETP Strategic Research Agenda (SRA) priorities should be established regarding:

— Core physics;
— Fuel data;
— Seismic studies;
— Instrumentation.

The project in support of the development of ASTRID, ALFRED, MYRRHA and ALLEGRO, with a large membership ranging from academia to industry, is supposed also to implement a large education and training program.

In addition to the co-financed so-called DG RTD programs, the DG JRC has a consistent direct research programme in support to GIF. In the recent JRC restructuring, all its nuclear activities are now merged in the Directorate for Nuclear Safety and Security. It is involved in many of the above-mentioned projects, including of course ESNII+.

Core safety calculations help to identify R&D needs for improving the core safety including multi-physics reactor performance, safety simulations and nuclear plant dynamics.

The fuel safety research involves the:

— Definition of state of the art fuel properties;
— Preparation of an adequate experimental programme;
— Characterization of fresh and irradiated MOX fuels;
— Properties measurements on fresh and irradiated MOX;
— Catalog on MOX properties for fast reactors.

The fuel work includes in addition to the standard MOX fuel the nitride and carbide and also metallic fuels, but also the inert matrix fuel (CERCER, CERMET) options. In addition to the standard characterization programs, a major focus is also on the compatibility with the liquid coolants (sodium, lead and lead-bismuth eutectic). Also, in the EERA-JPNM, the JRC has a leading role in the work packages on advanced fuel and industrial application of structural materials.

Furthermore, the JRC Geel in Belgium makes significant contributions to relevant neutron capture, fission and inelastic-scattering cross sections data in the energy range of about 1 keV to 10 MeV and contributes thereby essentially to a worldwide standardization of evaluated nuclear data for harmonized safety assessments in nuclear energy (Fig. 4)

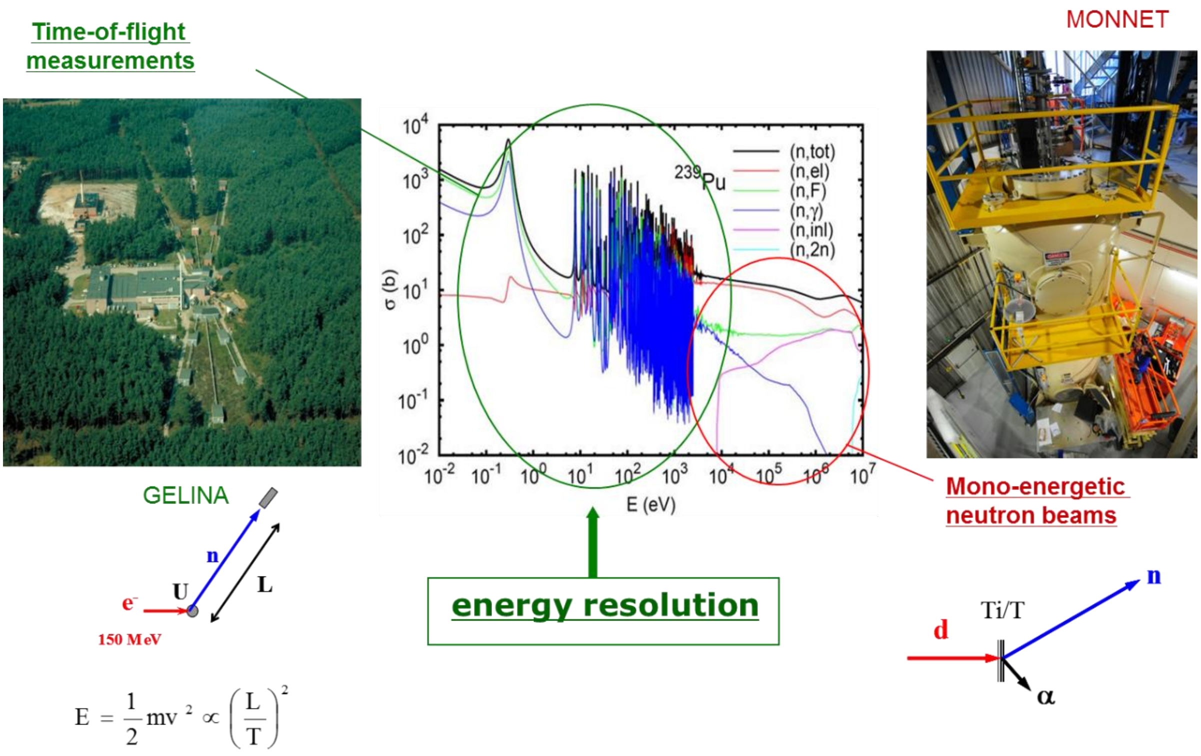

FIG. 4. Nuclear Data measurements at the JRC Geel, Belgium.

The unique nuclear research infrastructure is dedicated to the measurement of accurate nuclear reaction and decay data encompassing:

— a 150 MeV linear electron accelerator with a high-resolution neutron Time Of Flight (TOF) facility (GELINA);

— a continuous and pulsed proton, deuteron and helium ion beams facility which is serving as a source of well-characterized quasi-mono-energetic neutrons (MONNET);

— a broad set of experimental set-ups used for nuclear decay measurements, the radionuclide metrology laboratories (RADMET).

The measurements are essential of safe operation of nuclear reactors, especially fast reactors, wherein the view of a sustainable waste management the inventories of all long-lived actinides should allow a safe handling of nuclear waste and guarantee the radiological protection of the safety of the citizen and the environment.

Via the transnational access programme EUFRAT, JRC Geel offers external researchers from the EU Member States and third countries experimental possibilities at its nuclear facilities. The selection of experiments is based on a peer review process by international experts, representing the stakeholder community.

Based on its competences, the JRC is also strongly involved in methodology working groups namely:

— the Risk and Safety Working Groups (RSWS), were the JRC Petten plays a key role; and

— the Proliferation Resistance and Physical Protection Working Groups (PRPPWGs), where the JRC Ispra can rely on a large own expertise in the field of safeguards, proliferation and security.

The JRC is also a member of the cross-cutting task forces, the JRC Petten is a member of the SFR Safety Design Criteria (SDC) task force as shown in Fig. 5 [6].

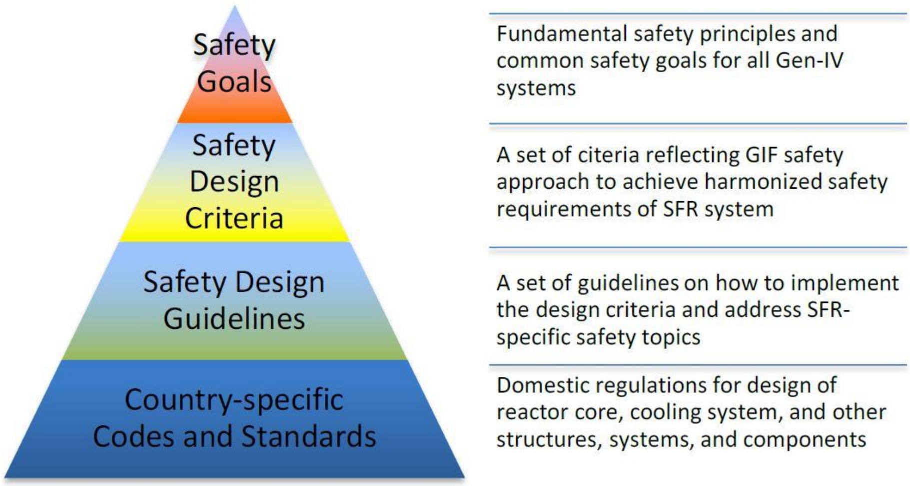

FIG. 5. SDC reference criteria for GIF safety design.

The JRC Karlsruhe represents EURATOM in the Education and Training Task Force (ETTF) a platform to enhance open education and training as well as communication and networking of people and organizations in support of GIF. The activities are connected to the European Nuclear Education Network, ENEN. The recently launched ENEN+ project financed in the frame of the Horizon2020 education and training programme has the main objective to attract, retain and develop new nuclear talents in view of new nuclear projects for the decades to come.

5. CONCLUSIONS

The SET-Plan includes nuclear in the energy mix in Europe, also to meet the CO_2 emission goals, i.e. to make an ongoing contribution to the decarbonization of the European energy system and to achieve the ultimate goal of reducing Europe's dependency on fossil fuels. Today in the EU there are 129 nuclear power reactors in operation in 14 Member States, with a total capacity of 120 GW(e) and an average age close to 30 years. The goal of the SNETP is to preserve and strengthen the European technological leadership and nuclear industry through a strong and long term R&D programme, involving fuel cycles and reactor systems of Gen II, III and IV. ESNII coordinates the Gen IV activities based on the reference SFR option with LFR and GFR as alternatives, including the R&D programs to foster the implementation of ESNII. The indirect and direct research programs in the European Commission support the EURATOM participation in GIF. DG RTD supports the R&D activities on the reactor systems, fuel and fuel cycle, advanced materials, codes and data with co-financed research programs. DG JRC is the implementing agent of EURATOM in GIF and participates in numerous of the above-mentioned programs including GIF system research and cross-cut activities. The JRC contributions in the field of nuclear data and in the frame of the EERA-JPNM project are essential and relevant.

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4.9. GENERATION IV INTERNATIONAL FORUM (GIF)

Overview of GIF Activities on Fast Reactor Systems

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1. BACKGROUND ON GENERATION IV INTERNATIONAL FORUM (GIF)

The Generation IV International Forum (GIF) was created in January 2000 by nine countries, and today has 14 members[1], all of which are signatories of the founding document, the GIF Charter[2], and 11 have signed the Framework Agreement to become active members.

GIF defined in its *Technology Roadmap* [1] four goal areas to advance nuclear energy into its next, "fourth" generation (see Fig. 1):

- Sustainability;
- Safety and reliability;
- Economic competitiveness;
- Proliferation resistance and physical protection.

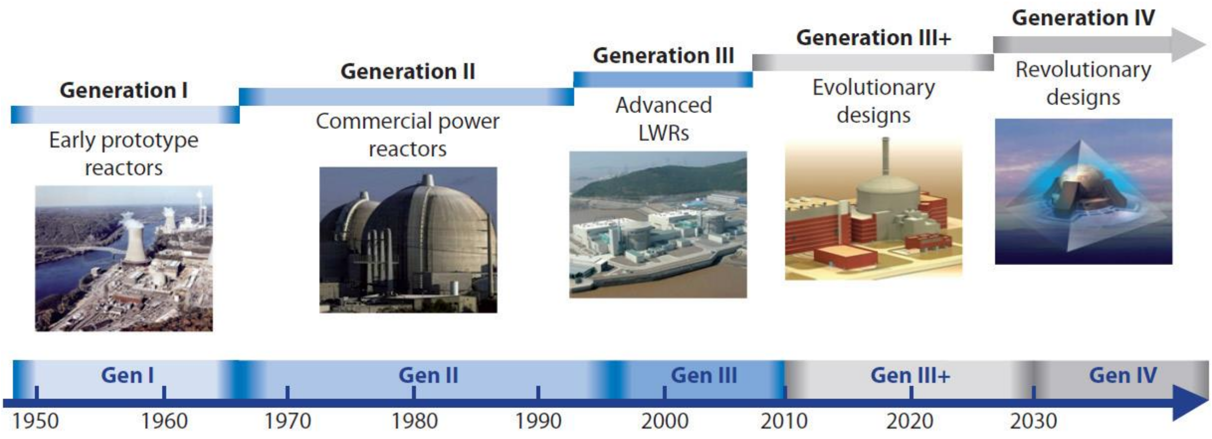

FIG. 1. The Four Generations of Reactor Designs.

The Technology Roadmap also defined and planned the necessary R&D to achieve these goals and allow for the deployment of Gen IV energy systems after 2030. Gen IV nuclear energy systems include the nuclear reactor and its energy conversion systems, as well as the necessary fuel cycle technologies.

[1]Argentina (non-active member), Australia, Brazil (non-active member), Canada, China, Euratom, France, Japan, the Republic of Korea, the Russian Federation, South Africa, Switzerland, the United Kingdom (non-active member) and the United States.

[2] The Charter was officially established in July 2001.

The closing of the nuclear fuel cycle is an important component for achieving the sustainability goal. It is based on the reprocessing and partitioning of spent nuclear fuel and the management of each fraction with the best possible strategy.

Fissile material, for example, can be recovered from the spent fuel and used to make new fuel. At present, almost 95 % of the spent fuel from light water reactors can be reused in the form of reprocessed uranium and MOX fuel.

With advanced fuel cycles using fast-spectrum reactors and extensive recycling, it may be possible to breed fissile fuel from fertile material and thus produce equal or more fissile material than the reactor consumes. This would also significantly reduce the footprint of deep geological repositories for the disposal of ultimate waste. The advanced separation technologies for Gen IV systems are being designed to avoid the separation of sensitive materials, and they include other features to enhance proliferation resistance and incorporate effective safeguards.

The *Technology Roadmap* [1] established an understanding of the ability of various reactors to be combined in so-called symbiotic fuel cycles, for example, through combinations of thermal reactors and fast reactors to accommodate transition periods. This was one of the primary motivations for having a portfolio of Gen IV systems rather than a single system in the original *Technology Roadmap* since various combinations of a few systems in the portfolio would provide a symbiotic system worldwide.

2. THE SIX GIF SYSTEMS

In 2002, a multi-criteria analysis was run GIF to identify the most promising concepts against the four goal areas previously mentioned (further refined in 15 criteria and 24 metrics).

Six systems were selected from nearly 100 concepts as Gen IV technologies:

- Gas cooled Fast Reactor (GFR) with a closed fuel cycle;
- Lead cooled Fast Reactor (LFR) with a closed fuel cycle;
- Molten Salt Reactor (MSR) with thermal and fast neutron concepts with a closed fuel cycle;
- Sodium cooled Fast Reactor (SFR) with a closed fuel cycle;
- Supercritical Water cooled Reactor (SCWR), theoretically with both fast and thermal neutron concepts (but designs with fast neutron spectrum are no longer investigated by GIF) with open or closed fuel cycle; and
- Very high temperature reactor (VHTR) with thermal neutrons and an open fuel cycle.

The GIF is clearly recognizing the major potential of fast neutron systems with closed fuel cycle. The 2014 update of *Technology Roadmap* [1] has confirmed the choice of these six systems.

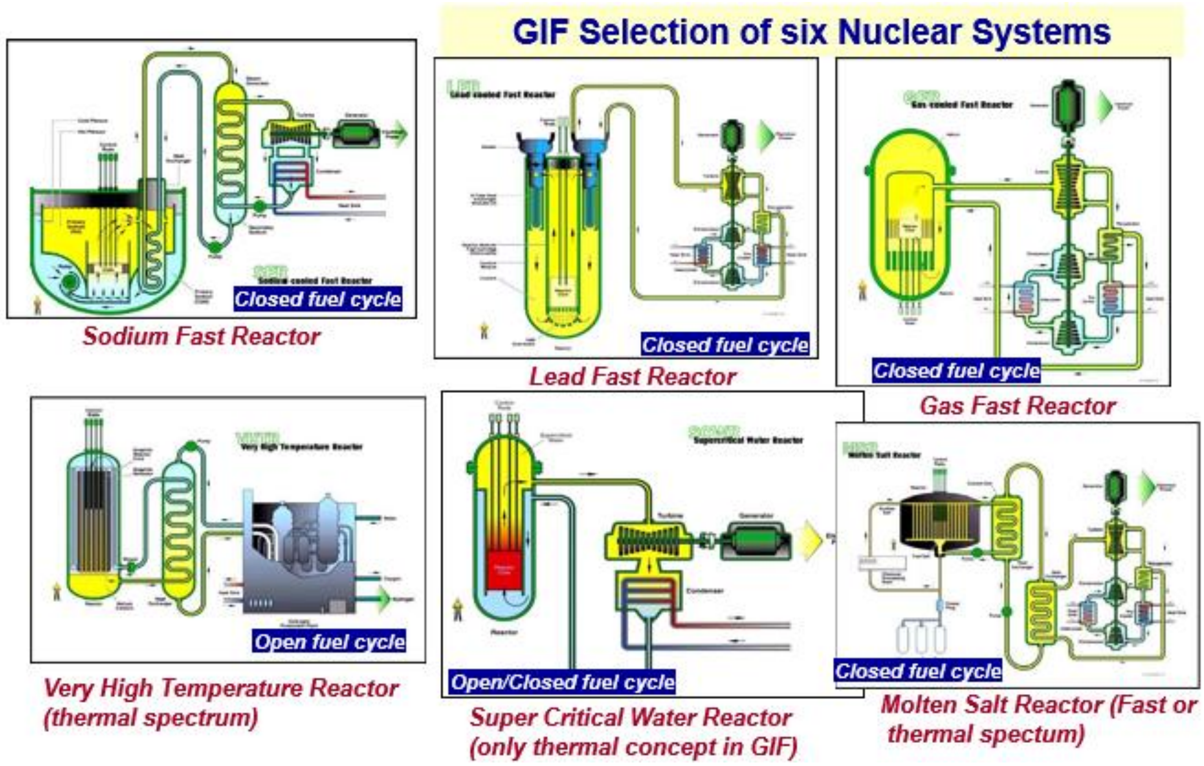

FIG. 2. The six GIF systems.

Timelines and research needs were developed for each system, categorized in three successive phases:

- The *viability phase*, when basic concepts are tested under relevant conditions and all potential technical show-stoppers are identified and resolved;
- The *performance phase*, when engineering-scale processes, phenomena and materials capabilities are verified and optimized under prototypical conditions;
- The *demonstration phase*, when detailed design is completed and licensing, construction and operation of the system are carried out, with the aim of bringing it to the commercial deployment stage.

This paper is aiming at reviewing the main R&D challenges and development status for the four GIF systems with a fast neutron spectrum (SFR, LFR, GFR and MSR).

3. DEVELOPMENT STATUS OF THE 4 GIF FAST NEUTRON SYSTEMS

3.1. Sodium cooled Fast Reactor concept

The status is the following:

- Three baseline concepts are investigated, loop configuration (Japanese JSFR design), pool-configurations (Korean KALIMER design, European ESFR design) and small modular SFR configuration (AFR-100 US design);

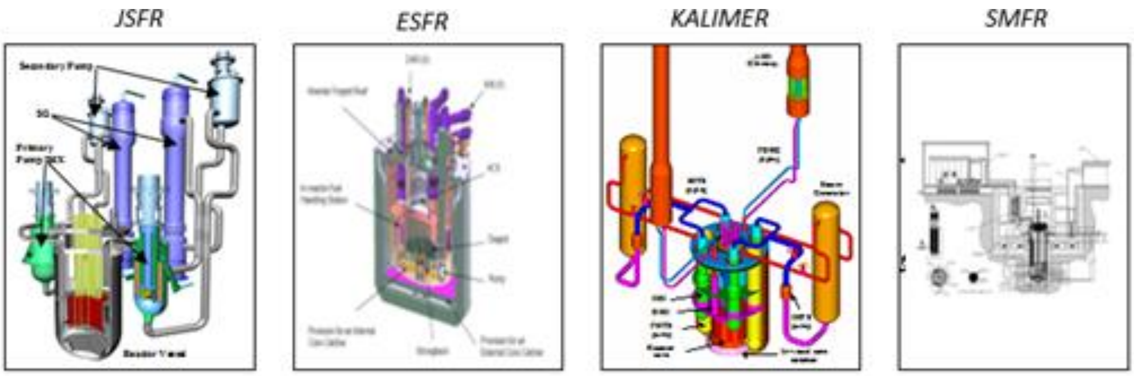

FIG. 3. The SFR Base-Line Concepts.

- Seven GIF members involved are China, EURATOM, France, Japan, Russian Federation, Republic of Korea and USA;
- Several SFR are operating or under construction (in China, India, Japan, Russian Federation).

The SFR system is probably the most mature system among the 6 GIF concepts. Approximately, twenty prototypes or demonstrators have been built throughout the world and they total more than 400 reactor-years of operation in 2012, as shown in Table 1 [2]. We are very pleased to have the chance to visit the BN-600 and BN-800 during the FR17 conference.

TABLE 1. WORLD FLEET OF SFRS AND TOTAL OPERATING DURATION – SITUATION IN 2012

Reactor (country)	Thermal Power MW(th)	Start (year)	Shutdown (year)	In Operation (years)
ERB-I (USA)	1.4	1951	1963	12
BR-5/BR-10 (Russia)	8	1958	2002	44
DFR (England)	60	1959	1977	18
EBR-II (USA)	62.5	1961	1944	33
FERMI 1 (USA)	200	1963	1972	9
RAPSODIE (France)	40	1967	1983	16
SEFOR (USA)	20	1969	1972	3
BN-350 (Kazakhstan)	750	1972	1999	27
PHENIX (France)	563	1973	2009	36
PFR (England)	650	1974	1994	20
KNK-II (Germany)	58	1977	1991	14
FFTF (USA)	400	1980	1993	13
SUPERPHENIX (France)	3000	1985	1997	12
JOYO (Japan)	50-75/100/140	1977		32
MONJU (Japan)	714	1994		15
BOR-60 (Russia)	55	1968		43
BN-600 (Russia)	1470	1980		31
FBTR (India)	40	1985		25
CEFR (China)	65	2010		1
BN-800 (Russia)	2100	Under Construction		
PFBR (India)	1250	Under Construction		
Total				40

- Most Gen IV SFR projects launched in GIF member countries are in performance phase (2012-2022) and will enter soon in demo phase for near term deployment (FOAK or industrial demonstrator);
- R&D efforts are concentrated on:
  - Safety & operation (core inherent safety, prevention /mitigation of severe accidents, minimization of sodium risks, ultimate heat sink, ISI&R, modularity, water/gas PCS);
  - Advanced fuel and TRU recycling capabilities;
  - Component Design and Balance of Plant for enhanced economics.
- Private companies (GE Hitachi, TerraPower, etc.) have demonstrated interest for GIF activities;
- Consolidation of common safety criteria for SFR systems draft safety design criteria and guidelines (SDC and SDGs) for SFR systems are now available (see dedicated panel, and reports available on GIF dedicated web-page [3]), together with the SFR

system safety assessment and the white paper addressing the main SFR safety features as evaluated using the integrated safety assessment methodology (ISAM) as elaborated by the GIF risk and safety working group (RSWG, see dedicated GIF web-page [4];

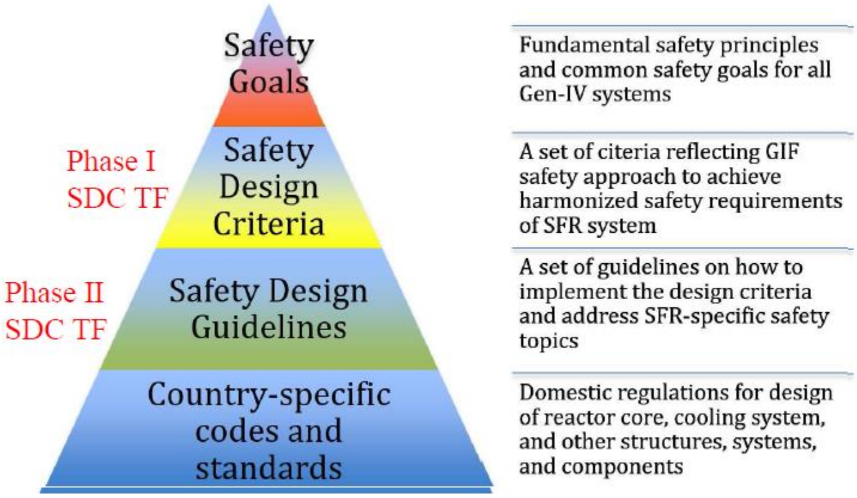

FIG. 4. GIF safety related activities (by the SDC/SDG Task Force and RSWG).

— Several webinars dealing with the SFR reactor technology have been assembled by the GIF Education and Training Task Force and are now available on the GIF dedicated web-page [5] (background on SFR technology, metallic fuels for SFRs, feedback experience from Phenix and Superphenix reactors, and a webinar scheduled in 2018 on Russia BN-600 and BN-800 reactors).

3.2. Lead cooled Fast Reactor concept

The status is the following:

— Lead cooled Fast Reactors (LFRs) are cooled by molten lead (or lead based alloys), a rather inert coolant (no rapid chemical reactions with water and air as it is the case for sodium) operating at high temperature and at near atmospheric pressure, conditions enabled because of the very high boiling point of the coolant (up to 1743°C) and its low vapour pressure. The coolant is either pure lead or an alloy of lead, most commonly the eutectic mixture of lead and bismuth, also known as LBE. The predominant coolant considered in the Gen IV reference LFR systems is pure lead;

— Three reference systems are considered as shown in Fig. 5. European large power reactor, 600 MW(e) (ELFR), an intermediate size (300 MW(e)) Russian reactor design (BREST-300), and a small transportable system of 10-100 MW(e) size (SSTAR) that features a very long core life-time;

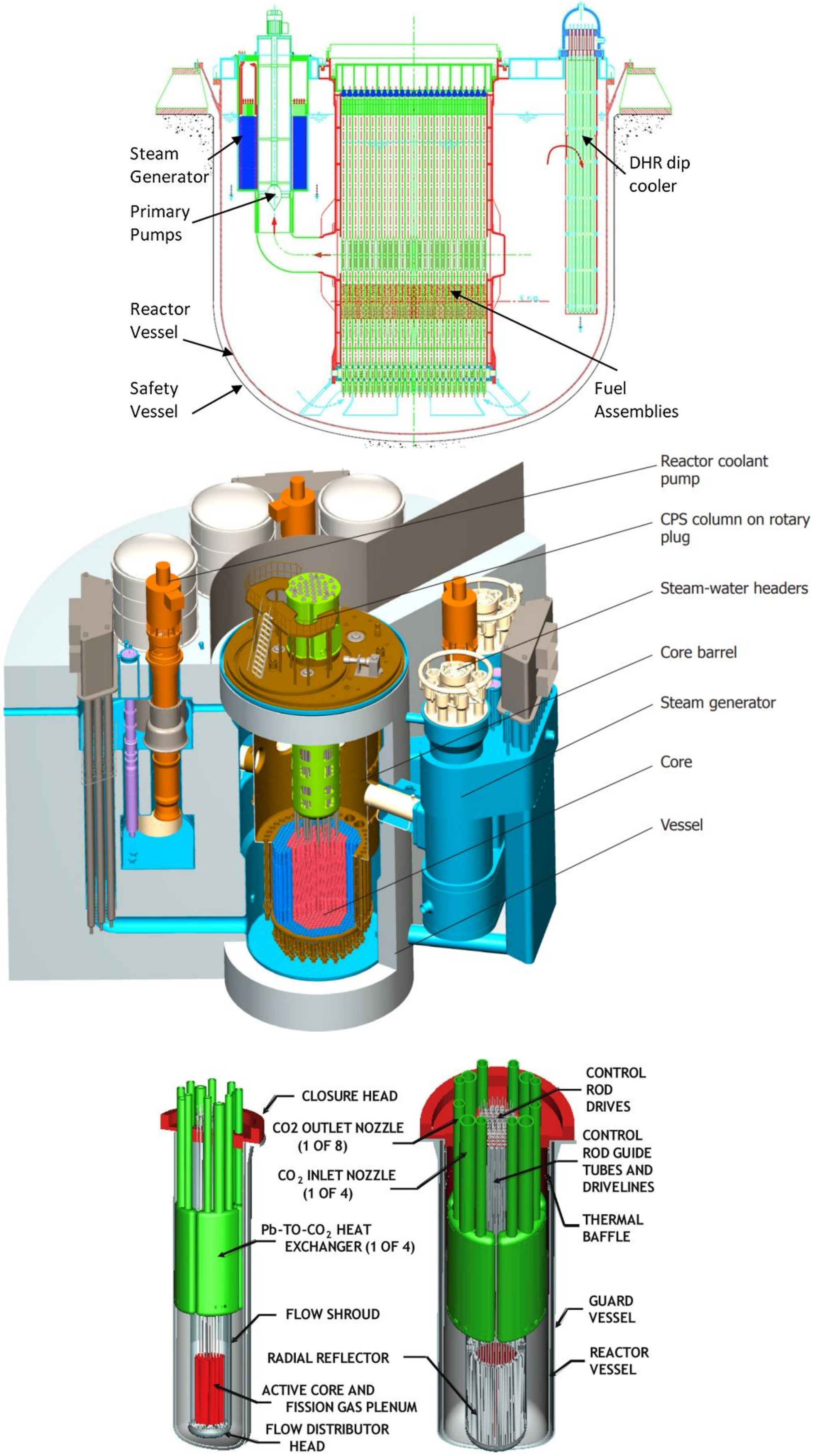

FIG. 5. Sketches of GIF-LFR reference systems (ELFR, BREST, SSTAR).

- Four GIF member Countries have signed the MoU setting the R&D cooperation framework (EURATOM, the Russian Federation, Japan, Republic of Korea, and Japan) and two (with the status of observers) have expressed their intention to join (US and China);
- One should also mention other important LFR projects worldwide with specific features such as the SVBR-100 Russian design using LBE as coolant (with a lower melting temperature ~ 124 °C in contrast with the 327°C of pure lead) or the ALFRED European project (investigating an LFR demonstrator with SMR oriented features), CLEAR 1 (10 MW(th) LBE ADS) and M (SMR design) in China. Lead technology is also driven by projects launched to study Accelerator-Driven Systems (MYRRHA for instance in Europe) using lead as coolant and target (to produce neutrons using spallation proton-induced reactions on lead);
- The main investigated R&D challenges (mainly due to the density and opacity of lead) are material corrosion, core instrumentation, in-service inspection and repair, fuel handling technology and operation, weight of primary system (seismic behaviour, sloshing, etc…), advanced fuel (dense nitride fuel, TRU recycling);
- The main operation experience is coming from LBE cooled reactors designed and built in the Soviet Union for the purpose of submarine propulsion. From the early 1960s until decommissioning of the final submarine in 1995, a total of 15 reactor cores were operated, providing an estimated 80 reactor years of operating experience [6]. The system has entered in the so-called performance phase in 2013 and is estimated to last for about 10 years with demo-reactor projects in the Russian Federation, Europe, Republic of Korea;
- Private companies (such as Westinghouse, Hydromine, or LeadCold) have also developed some basic LFR designs and have demonstrated interest in GIF activities;
- The GIF Risk and Safety Working Group has also published a “white paper” on LFR systems. This document includes a summary of the existing common features of the three LFR reference systems (ELFR, BREST, SSTAR), and then presents an application of ISAM methodology to the ALFRED demonstrator since, for this system, a consistent set of information has been disclosed and is available for the application. One should also mention that work on safety design criteria and guidelines (SDC and SDG) for LFR systems has just started;
- Other relevant events to quote on LFR systems are:
    - GIF-LFR Webinar by Prof. Craig Smith was held on June 12, 2017 (accessible on the GIF web site);
    - A first global symposium (GLANST) on HLM technology scheduled in Seoul, 7-8 September 2017.

3.3. Gas cooled Fast Reactor concept

The status is the following:

- The GFR cooled by helium is proposed as a longer term alternative to sodium cooled fast reactors. This type of innovative nuclear system has several attractive features; the helium coolant is a single-phase coolant that is chemically inert, which does not dissociate or become activated, is transparent and while the coolant void coefficient is still positive, it is small and dominated by Doppler feedback. The reactor core has a relatively high-power density, offering the advantages of improved inspection and simplified coolant handling. The high core outlet temperature above 750°C, typically 800-850°C is an added value to the closed fuel cycle;

- The reference concept for GFR is a 2400 MW(th) plant operating with a core outlet temperature of 850°C enabling an indirect combined gas steam cycle to be driven via three intermediate heat exchangers;
- A necessary step in the development of a 2400 MW(th) commercial GFR is the establishment of an experimental demonstration reactor for qualification of the refractory fuel elements and for a full-scale demonstration of the GFR-specific safety systems. This demonstrator will be ALLEGRO; a 75 MW(th) reactor with the ability to operate with different core configurations starting from a "conventional" core featuring steel-cladded MOX fuelled pins through to the GFR all-ceramic fuel elements in the latter stages of operation;
- GIF members involved in this system are the EU ("V4G4" legal entity or consortium, gathering four research institutes from Poland, the Czech Republic, Hungary and the Slovak Republic), France, and Japan;
- Key R&D challenges are mentioned here. The high core outlet temperature places onerous demands on the capability of the fuel to operate continuously with the high-power density necessary for good neutron economics in a fast reactor core. So we need robust fuel (ceramics clad UPuC) and structural materials. This represents the biggest challenge in the development of the GFR system. The second significant challenge for GFR is ensuring Decay Heat Removal (DHR) in all anticipated operational and fault conditions. Important R&D effort is aiming at improving the design for safe management of LOCA including depressurization, with a robust DHR without external power supply;
- This system is still in viability phase and is not expected to enter in performance phase before 2022;
- General Atomics (US) is working on a GFR design (EM^2 project) and has expressed interest in GIF activities;
- A "White Paper" on the safety of GFR systems has also been published with an application of the ISAM methodology. Developed in the framework of the Euratom project SARGEN_IV, under the supervision of the GIF GFR System Steering Committee (SSC) and RSWG, the paper compiles information that has been generated within the project and has been collected in the GFR related Euratom projects, GCFR STREP and GoFastR;
- A webinar on GFR system by Dr. Vasile (CEA, France) is also available on the GIF website;
- One should also underline some commonalities regarding R&D challenges for GFR and VHTR systems (especially regarding the development of high-temperature resilient materials or innovative intermediate heat exchanges, helium blowers and valve technology, etc…).

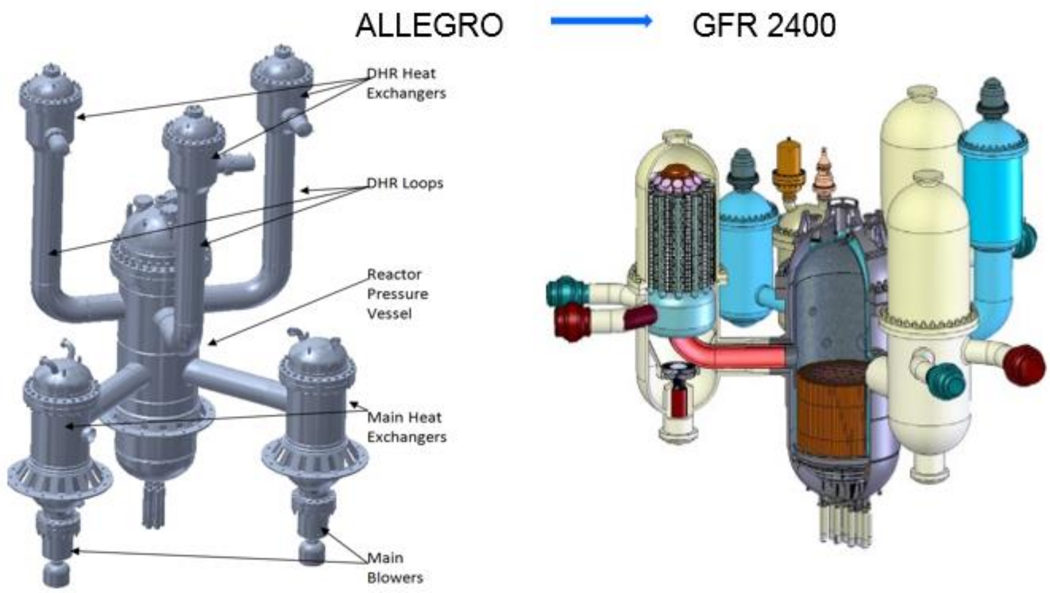

FIG. 6. GFR baseline concepts (ALLEGRO project, GFR-2400 MW(e) commercial reactor).

3.4. Molten Salt Reactor concept

The status is the following -

— Historically, this concept was developed in the fifties with molten salt used both as fuel and coolant, with graphite as moderator (reactor with a thermal neutron spectrum). Such liquid fueled reactors benefit from some potential advantages over solid fueled systems, among which are (i) The possibility of fuel composition (fertile/fissile) adjustment and fuel reprocessing without shutting down the reactor; (ii) The possibility of overcoming the difficulties of solid fuel fabrication/re-fabrication with large amounts of transuranic elements (TRUs); (iii) The potential for better resource utilization by achieving high fuel burnups (with TRUs remaining in the liquid fuel to undergo fission or transmutation to a fissile element);

  Since 2005, liquid fueled MSR R&D has focused on fast spectrum MSR options combining the generic advantages of fast neutron reactors (extended resource utilization, waste minimization) with those related to molten salt fluorides as both fluid fuel and coolant (low-pressure, high boiling temperature and, optical transparency);

— Nowadays, GIF MSR systems are divided into two main subclasses. In the first subclass (the only one existing in the 2002 GIF Technology roadmap), the fissile/fertile material is dissolved in the molten salt and it serves both as fuel and coolant in the primary circuit. In the second subclass (added in the 2014 update of the Technology roadmap), the molten salt serves as the coolant to a carbon moderated fuel similar to that employed in VHTRs;

— In order to distinguish the reactor types, the solid fuel variant is typically referred to as the FHR (Fluoride salt cooled High-Temperature Reactor) design initially investigated by the University of California Berkley. FHR concept is considered as a nearer term MSR option with a thermal neutron spectrum and a once through low-enrichment uranium fuel cycle;

— The two GIF baseline concepts for the liquid fuel options (with fast spectrum and closed fuel cycle) are the 1400 MW(e) MSFR design developed by France and

EURATOM within the SAMOFAR project (with a thorium fuel cycle) and the Russian 1000 MW(e) MOSART project (for actinide recycling/transmutation);

- Within the GIF PSSC-MSR (provisional System Steering Committee), research is performed on both subclasses, under an MOU signed by Euratom, France, the Russian Federation, Switzerland and the United States (with China, Japan, Australia and Republic of Korea as observers);
- Key R&D challenges are numerous for such a very innovative and prospective concept:
  - For the liquid-fuel option with closed fuel cycle, they are dealing with the salt properties (physical, chemical and thermodynamic properties) and solubility of actinides and fission products in the salt, system design and safety analysis (including development of advanced neutronic and thermal-hydraulic coupling models), development of advanced materials (including studies on their compatibility with molten salts and behaviour under high neutron fluxes at high temperature), corrosion and tritium release prevention based on proper molten salt Redox control, development of efficient techniques of gaseous fission products extraction from the fuel salt by He bubbling, fuel salt processing flowsheet (including reductive extraction tests for actinide/lanthanide separation), development of a safety-security approach (and proliferation resistance) dedicated to liquid-fueled reactors;
  - For the FHR solid-fuel option, specific R&D needs are to be addressed such as continuous fiber ceramic composites; FHR specific fuel elements and assemblies, etc.;
- Due to low TRL of MSR systems, the concept is still in feasibility or viability phase, with a performance phase expected to start by 2025;
- However, this concept seems to be very attractive for the private sector and national labs, with many projects launched worldwide:
  - In the US, both solid and liquid fuel options are investigated (FHR design, MCFR concept standing for Molten Chloride salt Fast Reactor, etc.). The GAIN initiative is offering the support of experimental means from National Labs. and NRC guidance for the licensing of various designs proposed by private companies (TerraPower, Thorcon, Terrestrial Energy, Flibe Energy, Transatomic Power, Elysium Industries, Alpha Tech. Research Corp., Kairos Power, …);
  - China is investigating two options, i.e. the FHR and the thorium Molten Fluoride Salt-thermal Reactor (TMSR) designs;
  - Other designs are proposed by Moltex (UK), Copenhagen Atomics (Denmark);
- The work aiming at testing the ISAM methodology on MSR system has been initiated;
- Two GIF webinars are addressing MSR systems, one on the FHR design option by Prof. Per. Peterson (UC, Berkeley, USA), the other one on the MSFR project by Dr. Elsa Merle (CNRS, France).

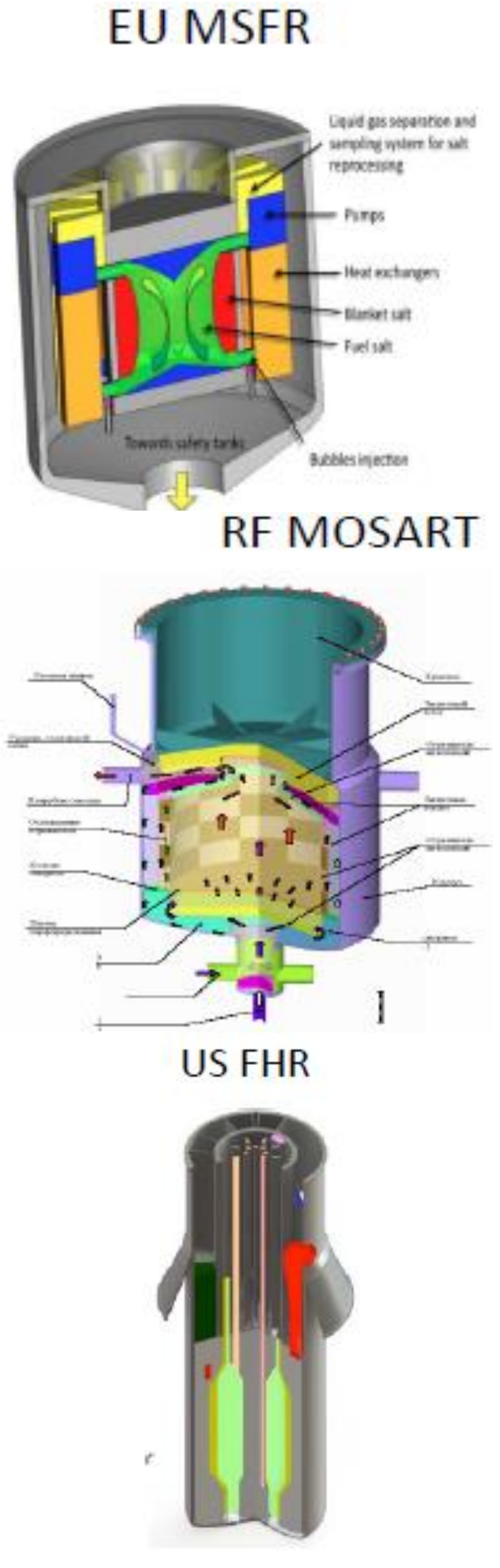

FIG. 7. Liquid (MSFR, MOSART) and solid (FHR) fuels MSR design-options.

4. CONCLUSION AND PERSPECTIVE

The 2014 update of the Technology Roadmap has confirmed the choice for the 6 most promising GIF reactor concepts with various TRL level and TRU recycling capabilities. Among the 11 GIF active member countries, 8 have demonstrated interest for fast reactor designs with closed fuel cycle. The SFR and LFR concepts are clearly the most advanced ones among the 4 fast spectrum GIF systems. The R&D on the GFR concepts has some obvious commonalities with the VHTR technology developments. Some private companies have shown interest for GIF research activities, for available experimental facilities among GIF member countries and GIF dedicated effort to develop a harmonized safety approach and licensing process. Pioneering work on safety design criteria and guidelines (SDC & SDG) has been completed for SFR systems. The GIF has also developed a powerful Integrated Safety Assessment Methodology (ISAM) that has been tested on the 4 GIF fast reactor systems. Moreover, a consistent set of webinar series is now available online through a dedicated GIF web-page (elaborated by the GIF Education and Training Task Force) introducing the main R&D challenges and lessons learned from feedback experience on the 6 GIF systems. GIF is currently working on an update of the 2009 R&D outlook to be presented at the 4th GIF symposium to be held in Paris (16-17 October 2018). This symposium is embedded in the 8th edition of Atoms for the Future and organized jointly by GIF and the French Nuclear Energy Society Young Generation Network (SFEN JG). MSc and PhD students, young professionals, policy makers and nuclear stakeholders are encouraged to participate in this symposium.

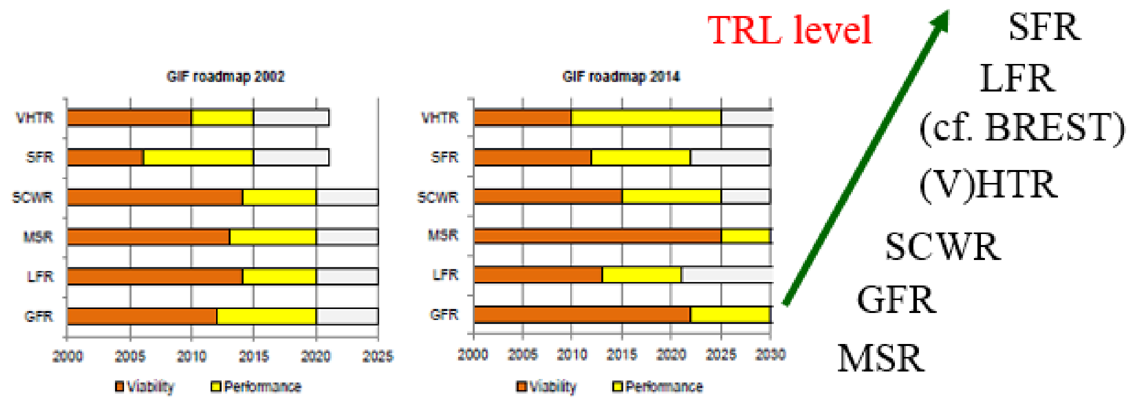

FIG. 8. TRL level for the 6 GIF systems (2014 update of the Technology Roadmap).

TABLE 2. OVERVIEW OF GIF MEMBER'S PARTICIPATION INTO THE 6 GEN IV CONCEPTS

	Canada (2001)	China (2006)	France (2001)	Japan (2001)	Rep. of Korea (2001)	Russia (2006)	RSA (2001)	Swiss (2002)	USA (2001)	EU (2003)
SFR		•	•	•	•	•			•	•
VHTR		•	•	•	•			•	•	•
LFR				•	•	•				•
SCWR	•	•		•		•				•
GFR			•	•						•
MSR*			•			•		•	•	•

(Year of Charter signed)

All activities, except LFR and MSR (based on MoU), are carried out based on a system arrangement.

Australia signed the Charter on 22 June 2016.

Argentina (2001), Brazil (2001), UK (2001). Non-active members (interest in UK for an active membership).

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4.10. OECD/NUCLEAR ENERGY AGENCY (NEA)

OECD Nuclear Energy Agency Activities Related to Fast Reactor Development

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Abstract. The mission of the OECD Nuclear Energy Agency (NEA) is to assist its Member Countries in maintaining and further developing, through international co-operation, the scientific, technological and legal bases required for a safe, environmentally sound and economical use of nuclear energy for peaceful purposes. It strives to provide authoritative assessments and to forge common understandings on key issues as input to government decisions on nuclear energy policy and to broader OECD analyses in areas such as energy and the sustainable development of low-carbon economies. This paper summarises the recent activities in the domain of fast reactors. Areas discussed include nuclear data for advanced reactors, integral experiments, efforts in knowledge preservation, uncertainty analysis and modelling of sodium cooled fast reactors, innovative fuels for fast reactors, identifying experimental needs for supporting fast reactor development, advanced fuel cycles, and work ongoing in the Generation IV International Forum (GIF). The NEA membership represents much of the world's experience in the area of fast reactors development and is an active forum to address the technological development of fast reactors.

Key Words: Experiments, Fast Reactor Development, Fuel Cycle, Innovative Fuel, Nuclear Data.

1. INTRODUCTION

The OECD Nuclear Energy Agency (NEA) is an international organization established to assist its Member Countries in developing the scientific, technological and legal bases required for the safe and economical use of nuclear energy for peaceful purposes. Within this mission, the NEA supports studies related to the development of fast reactor systems, covering both technical and strategic issues. This paper summarises recent and ongoing NEA activities in the field of fast neutron reactor system development.

2. NUCLEAR DATA FOR FAST REACTORS

Nuclear data represent a fundamental input needed to design and optimize nuclear reactors. At the NEA Nuclear Science Committee, nuclear data activities take place under the Working Party on International Nuclear Data Evaluation Co-operation (WPEC). WPEC was established in order to promote information exchange on nuclear data evaluations, measurements, nuclear model calculations, validation, and to provide a framework for co-operative activities between the participating projects. Within WPEC nuclear data improvements and needs are jointly assessed and collaborative efforts to improve the data are undertaken. Three WPEC activities have particular interest for the fast reactor community. They are described below.

2.1 Subgroup 39: Methods and approaches to provide feedback from nuclear and covariance data adjustment

During the previous Subgroup 33 "*Methods and issues for the combined use of integral experiments and covariance data*" it was pointed out that the statistical adjustments methodologies in use worldwide for different reactor analysis and design purposes are essentially equivalent and that they can provide a powerful tool for nuclear data improvement if used in an appropriate manner [1]. Covariance data, both those associated with nuclear data and those associated with integral experiments, play a crucial role in deriving calibrated, application specific libraries that can be used for fast reactor analysis.

Cross-section adjustment is increasingly perceived as providing useful feedback to evaluators and differential measurement experimentalists in order to improve the knowledge of neutron cross sections to be used in a wider range of applications. The adjustment has historically been used extensively to support fast reactor design.

Owing to this increase, the new role for cross-section adjustment requires tackling and solving a new series of issues: definition of criteria to assess the reliability and robustness of an adjustment; requisites to assure the quantitative validity of covariance data; criteria to alert for inconsistency between differential and integral data; definition of consistent approaches to use both adjusted data and a-posteriori covariance data to improve quantitatively nuclear data files; provide methods and define conditions to generalise the results of an adjustment in order to evaluate the extrapolability of the results of an adjustment to a different range of applications.

SG39 seeks to provide targeted feedback in order to improve future data files using synergies from different nuclear data projects. Within SG39 a review of issues and summary of methodologies used to provide feedback to evaluated data files (e.g. reactor physics experiment accuracies, adjustment methodologies etc.) have been performed.

Specific issue that was examined includes the consequences of using benchmarks, such as GODIVA and JEZEBEL [2] etc. that have been used implicitly in different evaluation projects. SG39 has examined how best to ensure convergence of recommended nuclear data adjustments and has studied best practices for avoiding large compensating adjustments among different isotopes, and ensuring the recommendations align with the knowledge of the differential data. The report scheduled for release in 2018 will present findings of this work, including agreed criteria for assessing robustness and reliability of an adjustment, criteria for the selection of integral experiments, an approach for validating a-priori covariance data, recommendations for the use of a-posteriori covariance data, a methodology and guidelines for providing feedback in order to improve neutron cross sections and associated covariance data in current evaluated nuclear data files, and practical applications to specific isotopes of priority interest for applications.

2.2 Subgroup 40: Pilot project of a Collaborative International Evaluated Library Organization (CIELO)

It is well known that the quality of the main evaluated data libraries is high and they tend to perform reasonably well in neutronics simulations for fission and fusion energy applications. However, not all users' needs in term of accuracy and completeness have been fulfilled. This is generally true for non-energy applications (e.g. accelerator, astrophysics) and for innovative energy applications (e.g. new energy range, new materials), but also for current fission and fusion systems (e.g. covariance data).

The nuclear data community has recognized that significant error compensations are still present in all files. For example, various ^{239}Pu fast neutron cross sections do not agree between the major nuclear data libraries outside the noted standard deviations contained in the files. Despite this, most tend to predict the proper critical mass for bare plutonium spheres such as JEZEBEL [2]. To better align the nuclear data evaluations, SG40 has created a wide international collaboration between major evaluated nuclear data libraries, to advance knowledge and understanding of the evaluation process and provide improved data for fission, fusion, and other nuclear applications, focusing on six important isotopes – ^{1}H, ^{16}O, ^{56}Fe, $^{235,238}U$, and ^{239}Pu.

Within SG40 participants have documented reasons for discrepancies between the existing main evaluated files, resolved some of the discrepancies, and advanced our understanding of the underlying cross sections and covariance's through advances in experiment, theory, simulation, and integral validation testing. This has resulted in high quality evaluations, parts of which have been adopted by the major nuclear data libraries. Additionally, a network of world experts to ensure that key differential and integral information is not omitted during the evaluation process. The result of this work will be issued at the beginning of 2018.

2.3 Subgroup 41: Improving nuclear data accuracy of ^{241}Am and ^{237}Np capture cross sections

The behaviour of minor actinides is important for the design of fast reactors, and for fuel cycle studies aimed at exploiting synergies between fast and thermal systems. Within WPEC SG-31 a state of the art review of experimental techniques for nuclear data measurements and the current status of nuclear data covariance evaluations was produced, where it was shown that at present there is still a serious gap between required accuracy and current accuracy [3]; required accuracies for fast reactor design were specified within WPEC SG26 [4]. Bridging this gap represents a major challenge for all file projects.

Therefore, SG41 is an international collaborative framework to improve the accuracy of evaluated data. Under this framework, all of the relevant forefront knowledge and techniques of energy dependent cross-section measurements, spectrum averaged experiments, nuclear data and associated covariance evaluations could be suitably integrated. In order to test the concept and assess the effectiveness of such a framework, SG41 focuses on two specific examples, i.e. the thermal and fast neutron capture cross sections of ^{237}Np and ^{241}Am. In the fast neutron spectrum significant uncertainty remains in these cross sections (see Fig. 1).

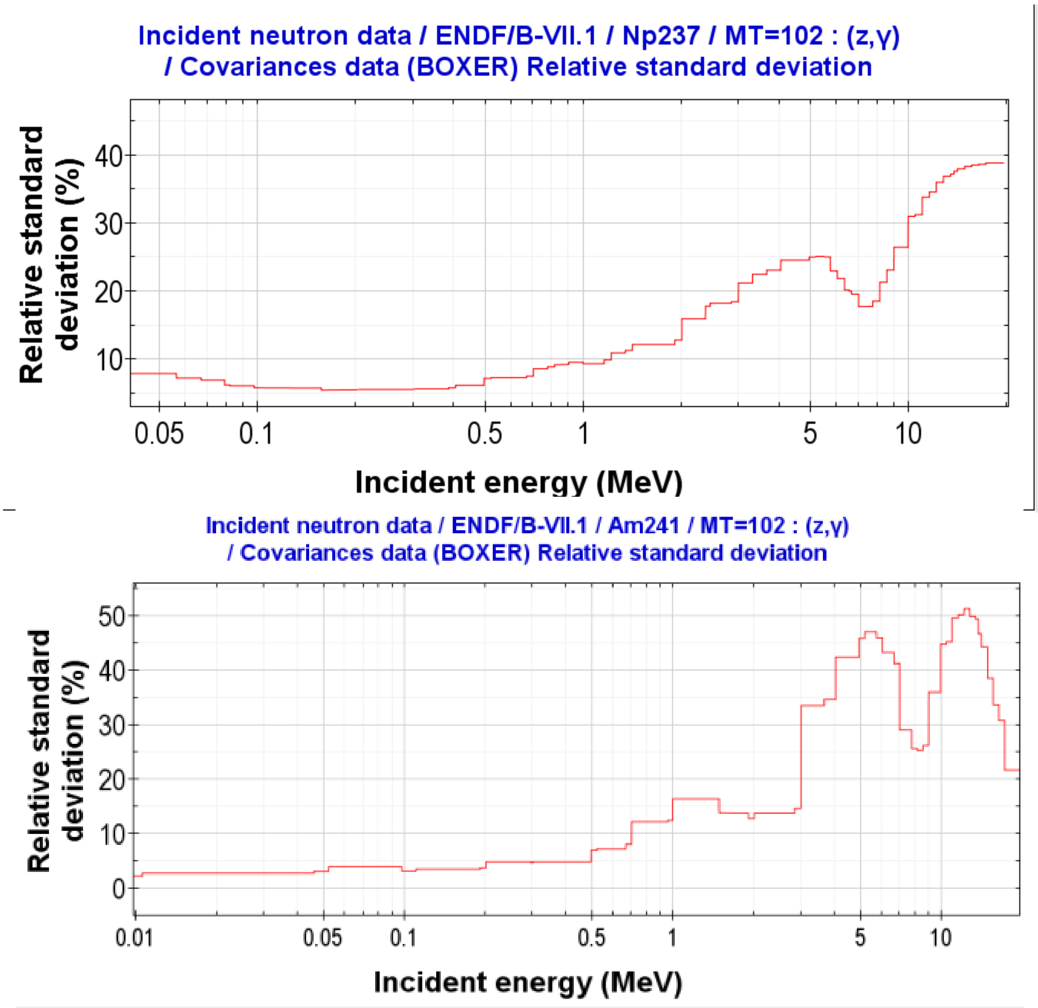

FIG. 1. Capture cross section uncertainties of ^{237}Np and ^{241}Am.

In order to design an international framework to improve the accuracy of evaluated data, all of the forefront knowledge of energy dependent cross section measurements, spectrum averaged experiments, relevant nuclear structure data, and evaluations are invested on two specific examples, i.e. the capture cross sections of ^{237}Np and ^{241}Am for thermal and fast neutrons. First, current evaluations are quantitatively assessed on these quantities. Second, the forefront knowledge on differential measurements was assessed, and the quantities are recommended based on the assessment. Third, the same was done using the forefront knowledge on spectrum averaged measurements. Fourth, the forefront knowledge on nuclear structure data measurements was assessed, and the relevant structure data are recommended. As the next step, the cross sections and covariance are to be updated by integrating all of the above assessments. The final report summarising results of this work is scheduled for 2019.

3. INTEGRAL EXPERIMENTS FOR FAST REACTORS

3.1 Criticality Experiments

The primary purpose of the International Criticality Safety Benchmark Evaluation Project (ICSBEP) Working Group is to compile critical and subcritical benchmark experiment data into a standardized format that allows criticality safety analysts to easily use the data to validate calculation tools and cross-section libraries. Currently, the ICSBEP Handbook [2] contains 686 benchmarks that have a fast neutron spectrum conducted in facilities such as BFS, ZPPR, ZEBRA, SNEAK and others.

ICSBEP work results in high-quality experimental benchmarks due to a rigorous evaluation process that includes:

— identifying a comprehensive set of critical benchmark data and, to the extent possible, verifying the data by reviewing original and subsequently revised documentation, and by talking to experimenters or individuals who are familiar with the experiments or the experimental facility;
— evaluating the data and quantifying overall uncertainties through various types of sensitivity analysis;
— compiling the data into a standardized format;
— performing calculations of each experiment with standard criticality safety codes;
— formally documenting the work into a single source of verified benchmark critical data.

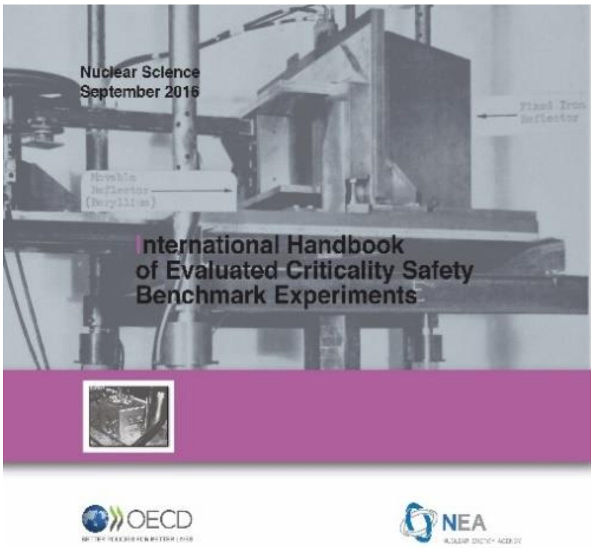

FIG. 2. Cover of the 2016 ICSBEP Handbook.

Every year, a new version of the ICSBEP Handbook is updated with dozens of benchmark configurations and released. A cover of the 2016 ICSBEP Handbook is shown in Fig. 2.

As the Handbook contains 4916 critical and subcritical assemblies, a Database for the ICSBEP (DICE) [5] was created to search and trend the data, as well as to store information such as sensitivity coefficients and the correlation coefficients between the integral experiments. DICE is accessible both via a DVD, or as a java webstart application. A screenshot of DICE is shown in Fig. 3.

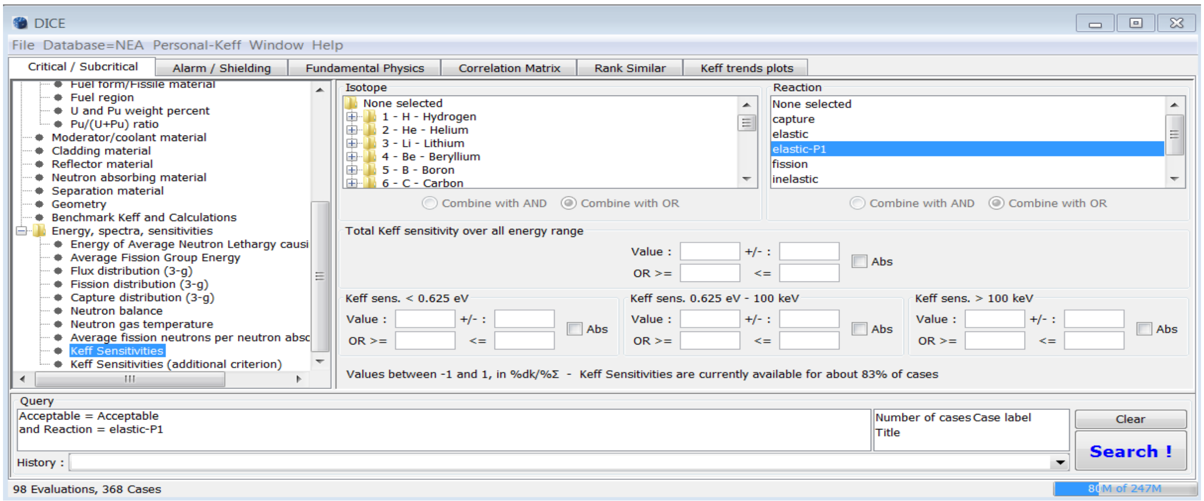

FIG. 3. Screenshot of the Database for ICSBEP (DICE).

3.2 Reactor Physics Experiments

The International Reactor Physics Experiment Evaluation (IRPhE) Project aims to provide the nuclear community with qualified benchmark data sets by collecting reactor physics experimental data from nuclear facilities worldwide. More specifically the objectives of the expert group are as follows:

- Maintaining an inventory of the experiments that have been carried out and documented;
- Archiving the primary documents and data released in computer-readable form;
- Promoting the use of the format and methods developed and seek to have them adopted as a standard;
- Compiling experiments into a standard international agreed format;
- Verifying the data, to the extent possible, by reviewing original and subsequently revised documentation, and by consulting with experimenters or individuals who are familiar with the experimenters or the experimental facility;
- Analysing and interpreting the experiments with current state of the art methods;
- Publishing the benchmark evaluations electronically.

Currently, the handbook [6] contains 25 Liquid Metal Fast Reactor Experiments performed in BFS, ZPPR, JOYO, FFTF and other facilities. Ongoing efforts to capture PFR data are described in a companion paper [7].

The expert group identifies gaps in data and provides guidance on priorities for future experiments. This community of practice has made an effort to involve the young generation (Masters and PhD students and young researchers) to find an effective way of transferring know-how in experimental techniques and analysis methods. Furthermore, the data represents the most complete set of experiments supporting Gen IV reactors.

The types of measurements included in the evaluations are Criticality, Buckling, Spectral Indices, Reactivity Worths, Reactivity Coefficients, Kinetics Parameters, Reaction Rate Distributions, Power Distributions, and Isotopic Measurements. The Handbook updated with new configurations is released yearly and can be requested from the NEA website. A cover of the 2016 IRPhEP Handbook is shown in Fig. 4.

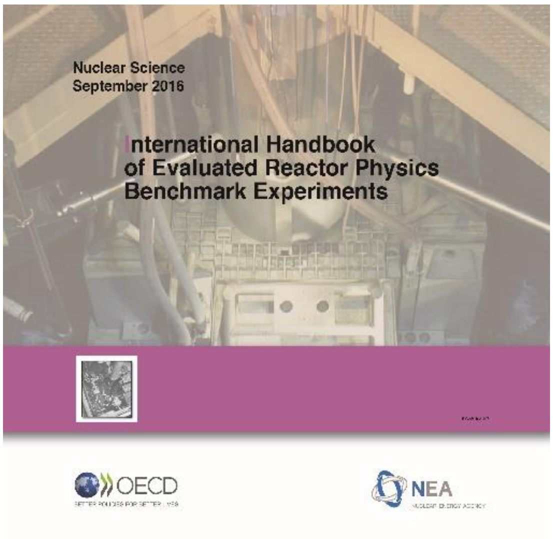

FIG. 4. Cover of the 2016 IRPhEP Handbook.

As the Handbook contains tens of thousands of pages, a database tool called the IRPhEP Database and Analysis Tool (IDAT) [8] was created to search and trend the data, as well as to store information such as sensitivity coefficients. IDAT is accessible both via a DVD or as a java webstart application. IDAT has been used to perform a rapid survey of the database contents, see Fig. 5, which shows the performance of various nuclear data libraries at predicting spectral indices of ^{237}Np fission/^{239}Pu fission.

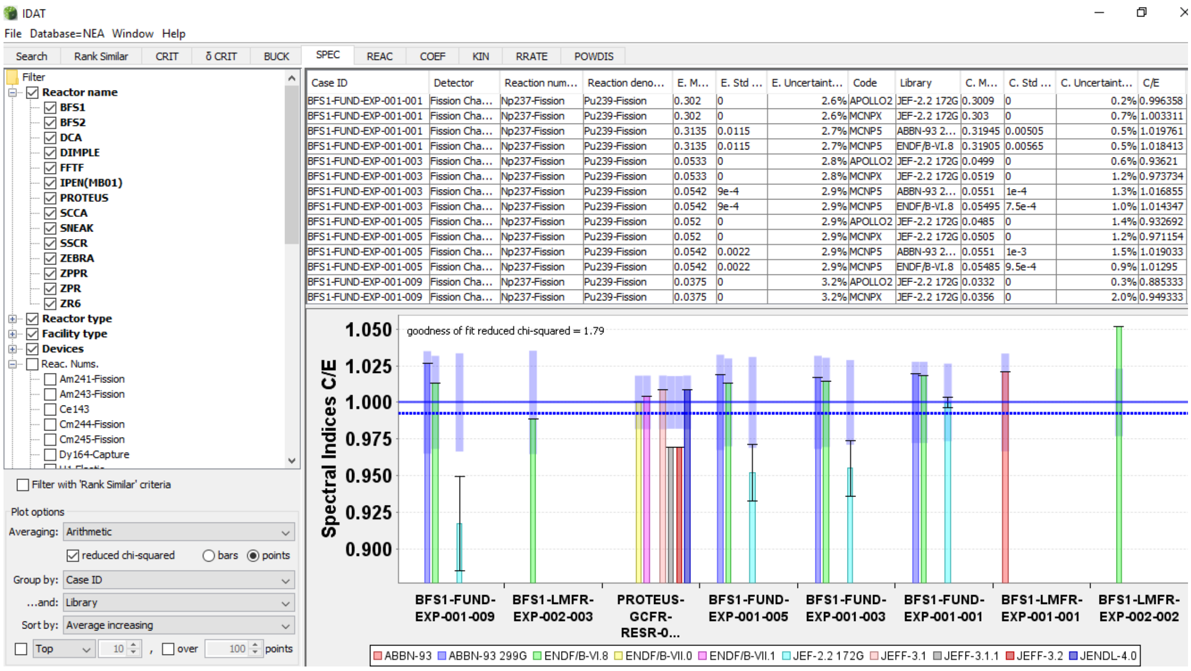

FIG. 5. Screenshot of the IRPhEP Database and Analysis Tool (IDAT).

3.3 Experiments for Minor Actinide Management

The Expert Group on Integral Experiments for Minor Actinide Management (EGIEMAM) reviewed the existing integral experiments for Minor Actinide (MA) management and identified a lack of experiments and insufficient accuracies in several areas [9]. In many cases, useful results are not fully available because of proprietary considerations. Moreover, EGIEMAM performed uncertainty analyses, target accuracy assessments and confirmed needs for improvement of nuclear data. After reviewing the integral experiments, EGIEMAM recognized that only a limited number of facilities and limited expertise and resources (materials, manpower and funding) exist.

Thus, the follow-up activity EGIEMAM-II was launched, there is a need to prepare a concerted effort paving the way for a common experimental programme where resources can be optimized towards improving the MA nuclear data knowledge. In conclusion, EGIEMAM has recommended integral measurements, complementary to parallel efforts for differential measurements, for the following nuclides of MA from viewpoints of design of transmutation systems and of fuel cycles: ^{237}Np, ^{241}Am, ^{242m}Am, ^{243}Am, ^{242}Cm, ^{243}Cm, ^{244}Cm and ^{245}Cm.

To improve knowledge of MA nuclear data and to support the MA management technology development with reliable accuracy and sufficient anticipation, the expert group pointed out that many additional integral data are still necessary. The first step in this direction requires pooling resources and identifying qualified facilities, personnel, measurement techniques and available supplies of materials to target experiments to meet specific MA data needs. From the lessons learnt, two major categories, reactor physics and irradiation experiments, require specific actions through international collaboration. The candidate facilities that can potentially be used for the collaborative experimental efforts as well as necessary measurements are shown in Fig. 6.

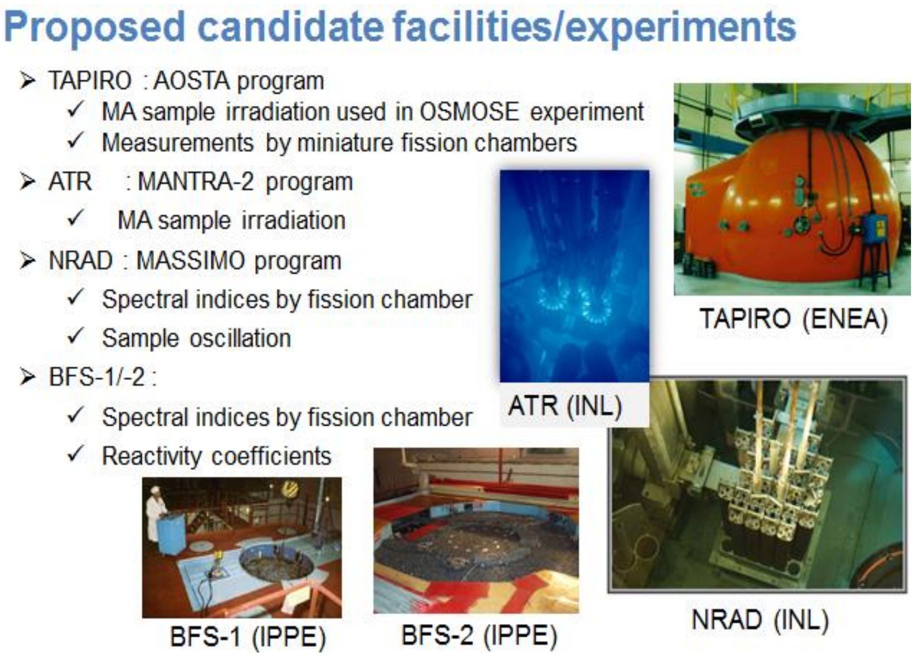

FIG. 6. Overview of integral experiments under consideration for Minor Actinide measurements.

Objectives of the continued efforts in this area undertaken by the EGIEMAM-II include, among others, the identification of systems of interest and associated target uncertainties, joint design of reactor physics MA measurements in selected facilities, and development and coordination of the irradiation programme (including assessment and sharing of resources and results, time schedule and cost). The findings of this activity will be documented in the report scheduled to be released in 2018.

4. UNCERTAINTY ANALYSIS IN MODELLING OF SODIUM COOLED FAST REACTORS

In recent years there has been an increasing demand for nuclear research, industry, safety and regulation for best estimate predictions to be provided with their confidence bounds. To address this demand the NSC Expert Group on Uncertainty Analysis in modelling has been created under the auspices of the Working Party on Scientific Issues of Reactor Systems (WPRS) with the objectives to elaborate a state of the art report on current status and needs of sensitivity and uncertainty analysis in modelling; to identify the opportunities for international cooperation in this area that would benefit from coordination by the NEA NSC and to draw a roadmap for the development and validation of the methods and codes required for uncertainty analysis including the benchmarks adequate to meet those ends, the schedule and organisation of its realisation.

The subgroup on Uncertainty Analysis in Modelling for Design, Operation and Safety Analysis of Sodium cooled Fast Reactors (SFR-UAM) has been formed under the EGUAM and is currently undertaking preliminary studies after having specified a series of benchmarks. Details of this work can be found in the companion paper [10].

5. ADVANCED FUEL CYCLES FOR FAST REACTORS

Activities in this area cover all aspects of the fuel cycle from the front end to the back end, and deal with issues arising from various existing and advanced systems including fuel cycle scenarios, innovative fuels and materials, separation chemistry and waste disposal and coolant technologies. To contribute to the sustainable development of nuclear energy, experts of the Working Party on Scientific Issues of the Fuel Cycle (WPFC) are currently focusing their work on improving nuclear fuel performance, developing materials, fuels and fuel cycles for new, innovative nuclear systems and managing spent fuel through reprocessing and recycling. The details of these activities are elaborated in the companion paper [11].

6. INTERNATIONAL FORUM ON ADVANCED REACTOR TECHNOLOGIES

For more than a decade, GIF has led international collaborative efforts to develop next generation nuclear energy systems that can help meet the world's future energy needs. Gen IV designs will use fuel more efficiently, reduce waste production, be economically competitive, and meet stringent standards of safety and proliferation resistance.

With these goals in mind, some 100 experts evaluated 130 reactor concepts before GIF selected six reactor technologies for further R&D. These include the Gas cooled Fast Reactor (GFR), Lead cooled Fast Reactor (LFR), Molten Salt Reactor (MSR), Supercritical Water cooled Reactor (SCWR), Sodium cooled Fast Reactor (SFR) and Very High Temperature Reactor (VHTR).

Eight technology goals have been defined for Gen IV systems in four broad areas, sustainability, economics, safety and reliability, and proliferation resistance and physical protection. These ambitious goals are shared by a large number of countries as they aim at responding to the economic, environmental and social requirements of the 21st century. They establish a framework and identify concrete targets for focusing GIF R&D efforts.

These goals guide the cooperative R&D efforts undertaken by GIF members. The challenges raised by GIF goals are intended to stimulate innovative R&D covering all technological aspects related to design and implementation of reactors, energy conversion systems, and fuel cycle facilities.

In light of the ambitious nature of the goals involved, international cooperation is considered essential for a timely progress in the development of Gen IV systems. This cooperation makes it possible to pursue multiple systems and technical options concurrently and avoid any premature down selection due to a lack of adequate resources at the national level.

7. ADVANCED REACTOR SYSTEMS AND FUTURE ENERGY MARKET NEEDS

It is clear that future nuclear systems will operate in an environment that will be very different from the electricity systems that accompanied the fast deployment of nuclear power plants in the 1970s and 1980s. As countries fulfill their commitment to decarbonize their energy systems, low-carbon sources of electricity and in particular variable renewables will take large shares of the overall generation capacities. This is challenging since in most cases, the timescale for nuclear technology development is far greater than the speed at which markets and policy/regulation frameworks can change.

An International Workshop on Advanced Reactor Systems and Future Energy Market Needs was organised by NEA in April 2017 to discuss how energy systems are evolving towards low-carbon systems, what the future of energy market needs is, the changing regulatory framework from both the point of view of safety requirements and environmental constraints, and how reactor developers are taking these into account in their designs. In terms of technology, the scope covered all advanced reactor systems under development today, including Gen IV fast systems.

8. CONCLUSIONS

The OECD NEA has a large number of activities taking place in the area of fast reactors. The breadth of activities spans from fundamental nuclear data to preservation of integral experiments performed on fast neutron systems, and from the safety assessment of fast systems all the way to a policy for national and international R&D. The NEA will continue to support member countries in the field of fast reactor development and related advanced fuel cycles, by providing a forum for the exchange of information and various other collaborative activities.

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4.11. INTERNATIONAL ATOMIC ENERGY AGENCY (IAEA)

Fast Reactors and Enhanced Nuclear Energy Sustainability

(Extended Abstract)

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Abstract. The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) promotes continuing dialogue and cooperation among the Member States in their respective roles as nuclear energy technology developers, suppliers and customers, since cooperation is an essential ingredient of sustainable development of nuclear energy. INPRO supports the Member States in their long term planning for sustainable nuclear energy. It provides this support through the direct provision of services on nuclear energy system scenario modelling, analysis and sustainability assessment using the INPRO Methodology.

INPRO was established through a General Conference resolution in 2000 to support efforts leading to the long term sustainability of nuclear energy so that it can help meet the rapidly growing energy needs of Member States in the 21st century and beyond [1]. This need has been highlighted by growing concerns about climate change, limited and unequally distributed energy resources and energy security. Nuclear energy is recognized as an option to effectively address these issues, but high capital investment, sophisticated human resources and complex institutional requirements implied by nuclear energy are not within reach for many nations to act unilaterally.

In response, INPRO brings together nuclear technology developers, suppliers and customers to jointly consider international and national actions, which could result in required innovations in nuclear reactors, fuel cycles and institutional approaches, to achieve viable and sustainable nuclear energy systems at national, regional and global levels. In so doing, Member States can move forward more deliberately in their strategic energy planning and decision making to help achieve important national and international outcomes.

INPRO has several top-level objectives that are central to its mission to help the Member States to develop national, regional and a global vision of nuclear energy sustainability.

- Seek shared visions of nuclear power development between the cooperating Member States
- Develop and provide a standardized analysis and assessment framework
- Engage the Member States in their respective roles as technology developers, suppliers and customers
- Provide venues and opportunities for participating Member States to leverage their resources (expertise, technologies, and infrastructure)
- Encourage communication and collaboration on nuclear energy system Research, Development And Deployment (RD&D)

INPRO is a membership-based project. Representatives of INPRO members form the INPRO Steering Committee (SC) that directly guides the project's activities. The INPRO Section, in the IAEA's Department of Nuclear Energy, coordinates activities with the Member States that have joined the project. The SC meets regularly to review progress and to provide guidance on future activities.

INPRO member countries include 41 IAEA Member States and the European Commission (EC): Algeria, Argentina, Armenia, Bangladesh, Belarus, Belgium, Brazil, Bulgaria, Canada, Chile, China, Czech Republic, Egypt, France, Germany, India, Indonesia, Israel, Italy, Japan, Jordan, Kazakhstan, Kenya, Republic of Korea, Malaysia, Mexico, Morocco, Netherlands, Pakistan, Poland, Romania, Russian Federation, Slovakia, South Africa, Spain, Switzerland, Thailand, Turkey, Ukraine, U.S., Vietnam and the EC.

1. INPRO METHODOLOGY FOR SUSTAINABILITY ASSESSMENT AND ITS APPLICATION TO FAST REACTORS

INPRO has two basic tool sets to consider issues of nuclear energy system sustainability. The first is the INPRO Methodology for sustainability assessment and the second is Nuclear Energy System (NES) modelling and simulation along with analysis and road mapping tools. The concept of INPRO focus is shown in Fig. 1.

The INPRO Methodology is holistic (multidimensional) NES sustainability assessment metric derived from the UN Brundtland Commission Report on sustainable development [2]. As in this foundational UN report, INPRO Methodology covers six main issues that directly affect nuclear energy sustainability, (i) proliferation, (ii) economics, (iii) health and environment risks, (iv) nuclear accident risks, (v) radioactive waste disposal, and (vi) sufficiency of national and international institutions. INPRO Methodology is a "self-assessment" performed by Member States' experts with assistance and review provided by Agency staff upon Member State request. Currently, INPRO Methodology is completing a 3rd update process since the first edition was published in 2003. As in the slide above, INPRO Methodology presents a static assessment of sustainability and seeks to discover possible "gaps" in a defined NES.

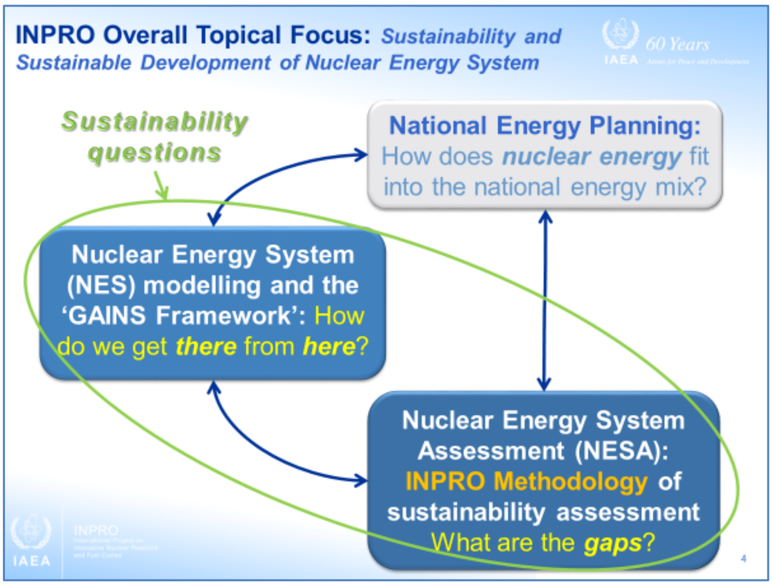

FIG. 1. INPRO focus on Sustainability and Sustainable Development of Nuclear Energy System.

Past applications of the INPRO Methodology to liquid metal fast reactor (and fuel cycle) included the so-called "Joint Study" completed in 2007 and finally published in 2013 as TECDOC-1639 Rev 1 and contains a generic assessment (no specific design basis) of closed fuel cycle and fast reactor using draft versions of the 2nd update of the INPRO Methodology, TECDOC-1575. TECDOC-1639 also contains a more detailed assessment of the Japan Sodium Fast Reactor (JSFR) based on an earlier version of the INPRO Methodology in TECDOC-1434 (2004). To date, INPRO (or the Member States) have yet to publish results of the application of the latest INPRO Methodology (TECDOC-1575) to specific innovative fast reactor designs.

More recently, three Member States undertook INPRO Methodology assessments of specific national fast reactor designs to demonstrate application innovative fast reactor designs and to check whether the Methodology could be reasonably applied to non-water cooled reactor designs (all previous cases of design specific assessments were to water cooled reactors). These assessments began in 2014 and have been completed in 2017. In each case, the Methodology was applied (design detail level) in two areas: economics and safety of nuclear reactor. India (IGCAR) assessed the CFBR (PFBR as reference plant), China (CIAE) assessed the CFR-1000 (CEFR as reference plant), and the Russian Federation (IPPE, OKBM) assessed the BN-1200 (BN-800 as reference plant). Currently, all the technical assessments have been performed and country reports are being edited following the requested INPRO review. India, China and the Russian Federation are currently considering if and how the results may be published and disseminated.

2. INPRO COLLABORATIVE PROJECTS ON FAST REACTORS AND ENHANCED NUCLEAR ENERGY SUSTAINABILITY

For the past several years, INPRO has pursued a set of "Collaborative Projects", with interested Member States, that have resulted in the development of a set of tools to perform modelling and simulation of NES, including the effects of fast reactors and cooperation among countries participating in potential fuel services trade relationships. Tools include the modelling and simulation tool (MESSAGE-NES), a key-indicators multi-criteria decision analysis tool (KIND-MCDA), and a road-mapping tool (ROADMAPS Template). These tools may be used together or separately to model and consider advantages and disadvantages of different potential sustainable development pathways of NES that may include fast reactors. As depicted in the Fig. 1 above, they seek to address the question of "how to get there from here". These tools, when combined with the application of INPRO Methodology for more detailed technological assessments addresses a full range of sustainability issues in NES planning studies.

While developing these tool sets, a generic scheme for "enhancing sustainability via advanced reactors and fuel cycles" was developed. The generic scheme is (i) structured along the lines of generic fuel cycle options, (ii) composed in a manner that presents a diversity of options that are inclusive of all Member States related policy positions (including nuclear phase-out policies), (iii) treats all options equally in a neutral manner, (iv) includes several options in which fast reactors may play a pivotal role, and (v) accepts the existing heterogeneous nature of technologies, infrastructures and policies among Member States and how the generic options may be realized through increased cooperation among countries to enhance the benefits received by a larger number of Member States through trade. The generic scheme is presented as a set of "options":

- Option A: Once-through nuclear fuel cycle;
- Option B: Recycle of spent fuel with physical processing (e.g., DUPIC and comparable);
- Option C: Limited recycling of spent fuel with chemical processing (e.g., mono-recycle of MOX in LWR using PUREX);
- Option D: Complete recycle of spent fuel U+Pu and possibly Np;
- Option E: Other minor actinides and fission product transmutation;
- Option F: Final geologic disposal of all high active, long-lived wastes.

To be sustainable, each generic option must be amended by option F (final geologic disposal, e.g., A+F, B+F, C+F, etc.). In the case of phase-out policies, the final state is option F since all radioactive wastes, requiring geologic disposal, must be interred to reach a final sustainable end state of the phased-out NES. Through cooperation and trade various national options can be enhanced by effectively adding services from other options. This is currently done to varying degrees through single and multiple bilateral agreements for cooperation and multi-lateral agreements have long been demonstrated as effective in the case of EURATOM, the EU still enjoys the highest fraction of nuclear electricity generation in the world, in part because of its close and well-regulated common market (the European Atomic Energy Community) established under the EURATOM Treaty in 1957 [3].

Example calculations completed under these various Collaborative Projects have demonstrated the benefits of cooperation through fuel services trade to reduce and manage the accumulation of spent nuclear fuel while reducing enormous investments in technology and infrastructure development through the pooling of markets to increase effective economies of learning and scale. Other example calculations have illustrated the system characteristics of both low growth and high growth cases for fast reactor deployments.

3. CONCLUSIONS

INPRO has developed tool sets and continues to develop IAEA services to assist Member States to plan for long term sustainable NES development through:

- Application of the INPRO Methodology for assessment of sustainability, including specific fast reactor designs;
- Application of scenario modelling and decision analysis tools for NES involving fast reactors, related fuel cycles and enhanced nuclear energy sustainability.

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5. SUMMARY OF TECHNICAL SESSIONS

During the conference, the session chairs were requested to provide the summary of each session. The following section is a compilation of edited summaries provided by the chairpersons. The IAEA acknowledges and appreciates their contribution.

5.1. TRACK 1 – INNOVATION FAST REACTOR DESIGNS

5.1.1. Session 1.1. SFR Design and Development

Six presentations were given in Session 1.1 followed by a general discussion.

P. Pillai (India) reported on the advanced design features of MOX fuelled future Indian SFRs. When discussing specific capital cost reduction of the reactor assembly, the author noted that these goals were achieved as a result of optimization of vessel dimensions, use of new shielding materials and changes made to component handling. Mr Pillai also noted that commissioning of the PFBR 500 MW(e) reactor awaited official regulatory authorization.

H. Bell (USA) provided an overview of the US fast reactor technology R&D programme.

S. Shepelev (Russian Federation) reported on the development of the new generation power unit for the BN-1200 reactor. In subsequent discussions, the author mentioned that the new unit will use MOX fuel and will be designated as power unit 5 of the Beloyarsk nuclear power plant. Cost estimates undertaken indicated that the cost per kilowatt-hour is slightly higher than a conventional LWR.

H. Hayafune (Japan) spoke about the advanced sodium cooled fast reactor development with respect to GIF safety design criteria. In discussions, the author noted that a probabilistic safety analysis had been carried out and that the design of the reactor components had been improved, thereby allowing the reactor to meet its seismic safety requirements.

In his second paper, H. Hayafune summarized the current status of the GIF collaboration on the sodium cooled fast reactors.

T. Obara (Japan) reported on the feasibility of the burning wave fast reactor concept with rotational fuel shuffling. In subsequent discussions, he noted that the 300 GW·d/t burnup rate was ensured by reshuffling of FSAs in the reactor core, which also ensures a stable power density profile in the core.

TABLE 1. PRESENTATIONS FROM SESSION 1.1. – SFR DESIGN AND DEVELOPMENT

Chair: **J. Guidez** and **A. Staroverov**

Id	Presenter	Country	Title
CN245-300 PPT-300	P. Pillai (Invited)	India	Advanced Design Features of MOX Fuelled Future Indian SFRs
CN245-357 PPT-357	H. Bell (Invited)	USA	Overview of U.S. fast reactor technology development programme
CN245-402 PPT-402	S. Shepelev (Invited)	Russian Federation	Development of the new generation power unit with the BN-1200 reactor
CN245-158 PPT-158	H. Hayafune	Japan	Advanced sodium cooled fast reactor development regarding GIF safety design criteria
CN245-156 PPT-156	H. Hayafune	Japan	Current status of GIF collaborations on sodium cooled fast reactor system
CN245-51 PPT-51	T. Obara	Japan	Feasibility of Burning Wave Fast Reactor Concept with Rotational Fuel Shuffling

5.1.2. Session 1.2. SFR Design and Development – II

Five papers were presented in Session 1.2. on SFR Design and Development. The papers of this session included broad areas covering the lessons from the ongoing PFBR project, R&D on the safety of future SFR systems, engineering management of the ASTRID project and metal fuel characteristics with heterogeneous configuration.

H. Kamide (Japan) delivered the first presentation on current progress in the design and related research on SFRs in Japan. The focus of the paper was on safety precautions undertaken after the severe accident at Fukushima. The experimental study employed three kinds of experimental facilities, PHEASANT, PLANDTL-II and AtheNa-RV, in the investigation of thermal-hydraulic phenomena under natural circulation DHR conditions, including the situation of the core disruptive accident (CDA) and evaluation of the performance of DHRs, and these were covered in detail. The behaviour of the molten core in the late phase of the CDA tests carried out for optimization of CRGT design and the design conditions of the self-actuated shutdown system were also covered.

E. Abonneau (France) presented the ASTRID project, providing a progress report from conceptual design to basic design. After the six years conceptual design phase, the basic design phase is now in progress (2016-2019). The entire configuration of ASTRID has been re-analyzed by taking into account feedback from the conceptual design phase towards reducing costs and towards simplification of systems. During consolidation phase of the reassembly, it was decided to reinforce some aspects and simplify the corium core catcher and reactor pit layout, including the ex-vessel decay heat removal system and creep-fatigue life duration of ASTRID components. It has also been decided to investigate and integrate a gas power conversion system in the basic design configuration in the next two years and it would be raised to the level of water/steam power conversion system.

V. Rajan Babu (India) presented Lessons and Strategies from PFBR to Future Fast Breeder Reactors. Starting from civil construction, manufacturing of over-dimensional and precision machined components, installation, integration, up to commissioning and operation of all the mechanical, electrical and I&C systems, there were many challenges and experience that had been obtained from the PFBR project. The lessons learned in these domains were highlighted which form important input for the design of future FBRs in India.

J. M. Hamy (France) presented the status of ASTRID Nuclear Island Design and Future Trends. AREVA is responsible for the design studies for the whole nuclear island project. Mr Hamy explained the technical challenges faced while carrying out the deployment of the design process based on system engineering standards and the selection of adequate architectures and design justification for the various main systems and components, which comprise primary circuit, secondary loops, decay heat removal systems, fuel and component handling systems, I&C systems, electrical systems, general layout of nuclear island building, etc. In this process, advanced numerical simulations, large CAD models and virtual reality tools were employed. The system engineering approach would also be continued in future activities.

I. Drobyshev (Russian Federation) presented “Analysis of the Characteristics of the Fast Breeder Reactor with Metallic Fuel”. The author explained the details of suggested heterogeneous core configuration required for achieving a high breeding ratio and a required temperature feedback reactivity coefficient. The layout of the depleted metallic fuel is proposed at the bottom blanket region and at the top blanket region above the sodium cavity to achieve a high breeding rate. In addition, placement of the oxide fuel with the central thin layer made of metallic fuel in the core is proposed to provide a required level of temperature feedback. In the future, it is planned to continue research to determine the most appropriate heterogeneous layout of core in terms of safety and breeding in the closed fuel cycle.

TABLE 2. PRESENTATIONS FROM SESSION 1.2. – SFR DESIGN AND DEVELOPMENT: II

Chair: **P. Pillai** and **S. Shepelev**

Id	Presenter	Country	Title
CN245-298 PPT-298	H. Kamide (Invited)	Japan	Progress of Design and related Researches of Sodium cooled Fast Reactor in Japan
CN245-413 PPT-413	E. Abonneau (Invited)	France	ASTRID Project, from Conceptual to Basic Design: Progress status
CN245-522 PPT-522	V. Rajan Babu (Invited)	India	Lessons and strategies from PFBR to Future Fast Breeder Reactors
CN245-528 PPT-528	J. Hamy	France	Status of ASTRID Nuclear Island Design and Future Trends
CN245-188 PPT-188	I. Drobyshev	Russian Federation	Analysis of the Characteristics of the Fast Breeder Reactor with Metallic Fuel

5.1.3. Session 1.3. System Design and Validation

Session 1.3. covered the topic of system design and validation and included five papers, two from India and three from France. The topics presented included seismic qualification of the shutdown mechanism, design of fuel handling systems and detailing of the gas conversion system vis-a-vis the conventional Rankine cycle steam–water system. Details of the presentations are given below.

S. Raghupathy (India) covered the details of seismic qualification of the diverse safety rod drive mechanism for the prototype fast breeder reactor. Two independent and diverse shutdown systems are provided in the reactor and the diverse safety rod, along with its drive mechanism, forms part of the second shutdown system. Details of the experimental seismic qualification tests carried out on the mechanism in stagnant water were presented. Tests were carried out on a special test facility and excitation was affected using hydraulic actuators. The mechanism was supported at the top, similar to that in the reactor, and the diverse safety rod was supported at the bottom of a structure simulating the grid plate. Excitations were given at three locations, namely, the mechanism support, the subassembly support and the subassembly button location. The input time history derived from the theoretical seismic analysis was applied. The qualification tests included measuring the increase in the free fall time, the critical parameter governing the drop and insertion of the diverse safety rod in its bottom position. Tests were carried out on two seismic levels, namely, the Operation Base Earthquake (OBE) and Safe Shutdown Earthquake (SSE). The measured value of the increase in free fall time was 180 and 205 ms under OBE and SSE, respectively. The details of the test set-up, instrumentation and methodology of the test were also presented. It was indicated that adequate margin exists on the free fall time, ensuring positive safe shutdown of the reactor under seismic conditions.

D. Planq (France) covered the progress made in the gas Power Conversion System (PCS) development for the ASTRID reactor. The gas PCS was studied as an alternative to avoid the possible sodium–water reactions in the steam PCS. Though the steam PCS was selected as the reference option in the conceptual design phase ending 2015, the alternative gas PCS was studied in detail during 2016–2017 to bring it to a par with the steam PCS. The system uses the nitrogen Brayton thermodynamic cycle and includes sodium gas heat exchangers, heat recuperator coolers, in addition to the multistage gas turbine. The presentation covered the various stages of evolution of the design, beginning in 2012, and the steps taken to improve the gross thermal efficiency of the system to 38.3% by 2017. The improvements made to the turbine design up to 2017 were highlighted. The details of the important components in the system such as the sodium gas heat exchanger, recuperator and pre-coolers and intercoolers along with the system layout were covered and the main domains of qualification requirements were also discussed.

D. Barbier (France) presented the operational aspects of the gas PCS in the same session. Details of the methodology of operation of the GSS PCS for turbine power control, maintaining the sodium temperature at steam generator outlet, turbine trip, decay heat removal, normal start-up, shutdown, scram, operation at house load, frequency control and the means to maintain gas inventory in the system were presented. The ongoing studies to increase the flexibility and increase the net efficiency of the cycle and the additional thermohydraulic studies planned for addressing incidents were also highlighted. Overall, the two presentations brought out the salient technical details and results of ASTRID studies carried out for the gas PCS.

F. Dechlette (France) presentation covered the French viewpoint on the ASTRID fuel handling system. In-vessel handling is proposed using two rotatable plugs and one straight pull machine located in the small rotatable plug, along with a fixed offset arm type machine located in the large rotatable plug. For ex-vessel handling out of the reactor vessel, an A-frame loading/unloading lock is used. A significant highlight of the design is the provision of the external buffer zone in sodium, which is interfaced with the reactor vessel. This helps to facilitate transfers between the buffer zone and the reactor vessel during fuel handling, resulting in significant reduction in fuel handling time. The exchange of fresh/spent fuel to and from the buffer zone is envisaged during reactor operation and the presentation brought out the requirements of the leak-tight airlock to enable this operation, which is being studied. The presentation also covered the details of sodium cleaning of the handling flask used to transfer from buffer zone to washing pits and the R&D related to sodium washing and future confirmatory studies required to firm up the concepts chosen.

S. Raghupathy (India) covered the details of component handling systems for future fast breeder reactors (FBRs) (twin unit 600 MW(e) FBRs-1/2) in India. The presentation highlighted the fuel handling and special handling system details for FBRs-1/2 as compared to the prototype FBR (PFBR). The experiences gained during testing of the principal fuel handling machines of the PFBR, i.e. the transfer arm and the inclined fuel transfer machine were explained. The commonality, as well as significant differences of the handling equipment with respect to the PFBR, were discussed in the presentation. The concept of in-vessel storage at the periphery of the core and ex-vessel storage in the water pool is retained. The significance of the increase in offset arm length by ~200 m and the change in ex-vessel handling equipment as a replacement for the in-vessel transfer machine was explained. Details of a unique layout of component handling equipment in fuel and decontamination buildings, which facilitates sharing of the equipment between the twin units was also presented. This had resulted in significant economy, with savings of 46% in material consumption and a reduction in overnight cost by ~2%.

Overall, the presentations made during the session were both interesting and technically illuminating. The topics of the presentations summarized the current status and efforts made by France and India for the design and validation of certain key systems of current and future FBRs. The session was very lively with active discussion of the topics.

TABLE 3. PRESENTATIONS FROM SESSION 1.3. – SYSTEM DESIGN AND VALIDATION

Chair: **S. Raghupathy** and **S. Rukhlin**

Id	Presenter	Country	Title
CN245-344 PPT-344	S. Raghupathy (Invited)	India	Experimental seismic qualification of diverse safety rod and its drive mechanism of prototype fast breeder reactor
CN245-285 PPT-285	D. Plancq	France	Progress in the ASTRID Gas Power Conversion System development
CN245-395 PPT-395	F. Dechelette	France	ASTRID fuel handling route for the basic design
CN245-314 PPT-314	S. Raghupathy	India	Component handling system: PFBR and beyond
CN245-468 PPT-468	D. Barbier	France	Main operation procedures for ASTRID gas power conversion system

5.1.4. Session 1.4. Core and Design Features – I

Session 1.4. comprised four presentations including two presentations from OKBM Africantov devoted to BN-800 core design.

P. Dařílek (Slovakia) from VUJE addresses ALLEGRO core design.

H. Yu (Republic of Korea) from KAIST described core physics study on the gas modular reactor.

A. Kuznetsov (Russian Federation) delivered two presentations on BN-800 core design. They generated considerable discussions on fuel design, core configuration, cladding material and core transition from hybrid core to full MOX. This indicates considerable interest in BN-800 core design.

The presentations generated discussions on analytical methods, results of analysis and design concepts.

TABLE 4. PRESENTATIONS FROM SESSION 1.4. – CORE AND DESIGN FEATURES – I

Chair: **H. Hayafune** and **D. Klinov**

Id	Presenter	Country	Title
CN245-406 PPT-406	A. Kuznetsov	Russian Federation	Selection of a layout for the BN-800 reactor hybrid core
CN245-37 PPT-37	P. Dařílek	Slovakia	ALLEGRO Core Neutron Physics Studies
CN245-405 PPT-405	A. Kuznetsov	Russian Federation	BN-800 core with MOX fuel
CN245-275 PPT-275	H. Yu	Republic of Korea,	Physics Investigation of a Supercritical CO_2–cooled Micro-Modular Reactor (MMR) for Autonomous Load-Follow Operation

5.1.5. Session 1.5. LFR Design & Development

The status and progress of Lead cooled Fast Reactors (LFRs) presented by GIF, China, Luxembourg and the Russian Federation were discussed in Session 1.5.

A. Alemberti (Italy) presented a summary reviewing the status of Gen IV LFR activities. The presentation addressed the status of GIF LFR Memorandum of Understanding and collaborative achievements made by the LFR provisional system steering committee, including the development of the LFR system research plan, the LFR white paper on safety, the LFR system safety assessment paper as well as the LFR safety design criteria. There were discussions on the qualification plan of materials used in the LFR and the IAEA's role in supporting the activity.

L. Cinotti (Luxemburg) introduced the simplified concept of the LFR-AS-200, which is under development by Hydromine and ENEA. It was suggested that the high level of safety features and the economic benefits of the LFR-AS-200 could be achievable through simplification of the system, i.e. in-vessel transfer machine, upper core structure, shielding elements and so on. Several issues on the I&C concept and the feasibility of in-service inspection were raised and discussed during the session.

Q. Huang (China) reviewed the R&D status of LFRs in China. The design concept of the China Lead-based Reactor (CLEAR) and the experimental facilities used to investigate key technologies of the LFR were discussed.

V. Lemekhov (Russian Federation) summarized the status of the BREST-OD-300 reactor design. The design concept of the BREST reactor and validation activities were introduced, as well as discussion of the test facilities for the BREST reactor. There were also discussions on justification of the power capacity of BREST-OD-300, and on corrosion and oxygen control in the reactor.

TABLE 5. PRESENTATIONS FROM SESSION 1.5. – LFR DESIGN & DEVELOPMENT

Chair: **J. Yoo** and **V. Lemekhov**

Id	Presenter	Country	Title
CN245-65 PPT-65	A. Alemberti	Italy	Status of Gen IV Lead Fast Reactor Activities
CN245-140 PPT-140	L. Cinotti	Luxemburg	Simplification, the atout of LFR-AS-200
CN245-301 PPT-301	Q. Huang	China	Strategy and R&D status of China Lead-based Reactor
CN245-539 PPT-539	V. Lemekhov	Russian Federation	BREST OD-300 reactor facility development stages and justification

5.1.6. Session 1.6. Core and Design Features – II

In Session 1.6., four papers related to ‘Core and Design features’ were presented.

I. Zverev (Russian Federation) presented on ‘Monitoring of Technical Condition of the Core in BN-1200 Advanced Commercial Sodium Cooled Reactor’. The focus of the presentation was on monitoring of fuel cladding leak-tightness at an NPP by a set of respective online and offline systems according to the activity measured in the primary circuit process media. The operating experience with the cladding tightness monitoring systems (CTMSs) in the BN-600 and BN-800 reactors were included in the development of the cladding tightness monitoring systems for the BN-1200. This ensures that the above-mentioned systems will have high reliability and effectiveness when used in the BN-1200 reactor. The major innovation for the BN-1200 reactor is that a fundamentally new monitoring schematic and a unique design have been introduced for the Na-CTMS, which keeps the primary coolant within the reactor vessel boundaries. To verify the effectiveness and the reliability of the new Na-CTMS schematic, the operating experience with a similar system in the BN-600 reactor was analyzed, and preliminary numerical calculations was made. The results of the accomplished analysis and calculations confirm the overall feasibility of placing the process part of the Na-CTMS into the reactor in-vessel equipment.

C. Venard (France) presented on ‘ASTRID Core at the end of the Conceptual Design Phase’. The presentation elaborated the safety goals and performance targets of the ASTRID core, the layout of the core and behaviour of the core during normal and transient conditions. The performances of the CFV V4 core comply the ASTRID project requirements. The new control rods architecture improves the core behaviour during CRW transient. The Complementary Safety Devices (CSD) improve strongly the core natural behaviour during the unprotected loss of coolant transients.

E. Rodina (Russian Federation) presented on ‘Fundamental approaches to High-power Fast Reactor Core development’. The project ‘PRORYV’ is for developing a fast reactor (BN-1200) with the capacity of 1200 MW(e) with a lead coolant. The key features of the BN-1200 core are multi-reprocessing fuel operation, breeding factor close to one and a small margin of reactivity change while needing only depleted uranium feedstock for refuelling after reprocessing. The study of neutronic and thermal characteristics for the FR-1200 reactor showed that the reactor core configuration designs variants comply the void effect requirements, reactivity overshoot between refuellings ~ β_{eff}, fuel load minimization and average fuel burnup requirements. Under steady-state refuellings with the nuclear fuel content closed to the equilibrium composition, average burnup values of 12% h.a. could be achieved with a fuel cycle length equal to 3000 effective days (the interval between refuelling – 600 effective days). The reactivity shift related to the increase in burnup would not exceed ~β_{eff}.

S. Belov (Russian Federation) presented on ‘Specific Features of BN-1200 core in case of use of nitride or MOX Fuel’. The development of the BN-1200 reactor core should ensure the high economic performance of the fuel cycle i.e., long lifetime and operating cycle length (~ 1 year). For the advanced nuclear fuel cycle, a requirement has been established to ensure the technological support for the proliferation-resistant mode. Keeping this in mind, a high plutonium breeding ratio needs to be ensured directly in the reactor core (Core Breeding Ratio (CBR)). For BN-1200, mixed uranium-plutonium nitride fuel is being considered as the advanced high-density fuel that ensures a high CBR. As the backup option, a core is being developed based upon the technologically mastered MOX fuel that can ensure high burnup. A flattened reactor core is used with an upper sodium plenum and an upper absorber shield to reduce the sodium void reactivity effect to the level of β_{eff}. The fuel is used with the same plutonium enrichment so that the power distribution is stable in time. It is ensured that both fuel types (MOX and nitride fuel) can be used.

The papers of this session covered broad areas including the core and design features in the reactors BN-1200, ASTRID and FR-1200.

TABLE 6. PRESENTATIONS FROM SESSION 1.6. – CORE AND DESIGN FEATURES – II

Chair: **V. Rajan Babu** and **S. Belov**

Id	Presenter	Country	Title
CN245-414 PPT-414	I. Zverev	Russian Federation	Core condition monitoring in advanced commercial sodium BN-1200
CN245-288 PPT-288	C. Venard	France	The ASTRID core at the end of the conceptual design phase
CN245-20 PPT-20	E. Rodina	Russian Federation	Fundamental Approaches to High-power Fast Reactor Core Development
CN245-408 PPT-408	S. Belov	Russian Federation	Specific features of BN-1200 core in case of use of nitride or MOX fuel

5.1.7. Session 1.7. ADS and other Reactor Designs

Session 1.7. comprised of five presentations from Belgium, France, Hungary, Japan and the Russian Federation.

T. Sasa (JAEA, Japan) presented the study on the Accelerator Driven System (ADS) in J-PARC/JAEA. The latest post-Fukushima strategic energy policy of Japan expresses to enhance R&D to reduce the burden of long-lived nuclides in spent nuclear fuel using both fast reactors and ADS. JAEA proposed transmutation of Minor Actinides (MA) by ADS and plans to construct the Transmutation Experimental Facility (TEF) within the framework of the J-PARC project. The presentation mainly discussed the activities to realize the TEF.

R. Fernandez (Belgium) presented on the evolution of the primary system design of the MYRRHA facility. SCK•CEN is proposing to replace its ageing flagship facility, the Material Testing Reactor BR2, by a new flexible irradiation facility, MYRRHA. Considering the international and European needs, MYRRHA is conceived as an ADS-based flexible fast spectrum irradiation facility able to work in both sub-critical and critical mode. Additional requirements, safety issues and R&D findings triggered the evolution of the mechanical design of the reactor. This evolution led up to the continuous increase of the dimensions of the reactor, increasing the costs and complicating the transport. Currently, the focus is on obtaining a more compact design by studying innovative components as double-wall heat exchangers, fuel handling machines and alternative configurations. These elements will form the input for the next revision of the MYRRHA Primary System.

A. Balanin (Russian Federation) presented the physical and technical basics of the concept of competitive Gas cooled Fast Reactor (GFR) facility with core based on coated fuel microparticles. The development of the reactor facility with gas cooled fast breeder reactor BGR-1000 possesses advantages from a synthesis of the proven technological decisions of high-temperature and light-water reactors. In comparison with GFR concepts with traditional container-type fuel elements (fuel rods), the concept of BGR-1000 reactor with a fixed bed of coated fuel microparticles has the potential to eliminate essential radiation consequences of accidents due to inherent properties of fuel design. The concept ensures the additional level of protection from the proliferation of the fissile materials. It is rather difficult to reprocess coated particles by the traditional aqueous methods, especially with the dense carbide fuel. At the same time, the reprocessing of coated particles is possible by the advanced high-temperature non-aqueous methods, for example, the method of volatile gas fluorides. The main directions of future investigations are the sensitivity analysis that should reveal possible issues requiring further qualification.

J. Gado (Hungary) presented the ALLEGRO experimental Gas cooled fast reactor project (GFR). ALLEGRO is an experimental fast reactor cooled with helium being developed by the European V4G4 Consortium “V4G4 Centre of Excellence” of the nuclear research organizations of the Czech Republic, Hungary, Poland and Slovakia associated with CEA (France). Development of ALLEGRO is an important step on the way to the Gas cooled Fast Reactor, one of the six concepts selected by the Generation IV International Forum (GIF) and one of the three fast reactors supported by the European Sustainable Nuclear Energy Technology Platform. The main purpose of the facility is to develop: innovative refractory GFR fuels, GFR-related components and systems (helium technologies, fuel handling, etc.) and, a safety framework applicable to the specific characteristics of GFRs. The current status and the perspective steps of the design and safety studies and experimental work to demonstrate the safety & feasibility of ALLEGRO were presented and discussed.

D. Gerardin (France), presented ‘Design Evolutions of the Molten Salt Fast Reactor and discussed design optimization studies of the molten salt fast reactor (MSFR). The segmented design of the fuel circuit described was shown to be optimized to suppress the risk of fuel leakage while offering a compact geometry for the circuit. The emergency draining system (EDS) designed for the MSFR allows to recover the fuel salt in case of in-core anomalies due to gravitational draining. The preliminary thermal calculations in the draining tank were shown and displayed that the fuel and inert salt layer thicknesses can be chosen in order to cool the fuel effectively while keeping it at liquid state for up to one month and to limit the maximal temperature reached at the metallic wall. To account for the water-fuel salt interaction and to suppress a possible accident initiator, the design studies of the EDS tends to be oriented to coolants other than water, typically a gas cooling system, in collaboration between CNRS, KIT and EDF. This task is undertaken in 2017 in the frame of the SAMOFAR project.

TABLE 7. PRESENTATIONS FROM SESSION 1.7. – ADS AND OTHER REACTOR DESIGNS

Chair: **V. Rachkov** and **S. Monti**

Id	Presenter	Country	Title
CN245-282 PPT-282	T. Sasa	Japan	Study for Accelerator-driven System in J-PARC/JAEA
CN245-358 PPT-358	R. Fernandez	Belgium	The evolution of the primary system design of the MYRRHA facility
CN245-517 PPT-517	A. Balanin	Russian Federation	Physical and technical basics of the concept of a competitive gas cooled fast reactor facility with the core based on coated fuel microparticles
CN245-574 PPT-574	J. Gadó	Hungary	The ALLEGRO experimental Gas Cooled Fast Reactor Project
CN245-575 PPT-575	D. Gerardin	France	Design Evolutions of the Molten Salt Fast Reactor

5.1.8. Session 1.8. Innovative Reactor Designs

Five presentations were delivered and discussed in Session 1.8, two from the Russian Federation, two from Sweden, and one from Japan.

A. Sedov (Russian Federation) presented a concept of the supercritical water cooled reactor VVER-SCP operating in a fast neutron spectrum. Main goals of VVER-SCP are: (i) the possibility of operation of the reactor in a regime of self-provision by fuel in the closed cycle; and (ii) energy efficiency of NPP should be not less than 40-42%. One of VVER-SCP concepts is a variant of two-circuit NPP with the fast reactor, cooled by light-water steam at supercritical pressure – SCPS-600, with electrical power of 600 MW(e). A brief description of SCPS-600 reactor concept was presented and addressed some aspects of neutron physics, thermal hydraulics and reactor stability. The studies show a possibility of the creation of fast supercritical steam reactor, which can operate in a regime of self-provision of the reactor by its own secondary fuel in the equilibrium closed fuel cycle.

S. Bortot (Sweden) introduced a conceptual core design of a 600 MW(e) lead cooled, nitride-fueled fast reactor aimed at transmuting of minor actinides (MA). The analysis of the core transient behaviour following postulated accident initiators confirmed that the safety reference criteria are respected also when deviations from the nominal operating conditions occur, as cladding failure, fuel melting and nitride dissociation are prevented with fairly good margins. It was concluded that the cladding surface temperature appears to be the most critical parameter. Therefore, more accurate transient analyses were suggested in order to definitely assess the core safety performance and confirm its capability to survive severe accidents.

K. Arie (Japan) presented four-years progress of the core design of the innovative uranium-free Transuranium (TRU) Burning Fast Reactor Cycle. The most effective way to burn TRU is to use uranium-free TRU fuel since it does not produce any new TRU. A feasible uranium-free TRU metal fuel core was identified and studied. The other researches, such as pyroprocess of uranium-free metal fuel and fuel irradiation performances, are also in progress.

J. Wallenius (Sweden) presented a 3-10 MW(e) lead cooled fast reactor operating on 19.9% enriched UO_2 fuel. The SEALER reactor is designed for commercial production of electricity in communities and mining operations in the Canadian Arctic. The general technical concept of SEALER was discussed together with the plan for licensing this reactor in Canada. A preliminary assessment of heat production in the aforementioned communities indicated no immediate commercial incentive, considering the cost for establishing infrastructure for district heating. Over a period of a few decades, approximately 100 reactors may be deployed for commercial off-grid power production in Canada.

N. Maslov (Russian Federation) reported on improvements of the ‘inherent safety’ of the BN-600 reactor by using fuel assemblies with (U, Pu)C microfuel. Fuel assemblies with pellet MOX fuel and fuel rods are directly replaced by microspherical mixed (U, Pu)C-fuel. A comparison of neutron physics and thermal hydraulics characteristics of the innovation fuel assemblies with microspherical mixed (U, Pu)C-fuel and the traditional fuel assemblies with pellet MOX fuel and fuel rods was conducted. The calculation model was BN-800 reactor core with MOX fuel with a three-zone radial power density. Thanks to the microspherical carbide fuel, the inherent safety of the reactor increases in accidents with loss of coolant flow and introduction of positive reactivity because the coated particles develop heat-exchange surface and their coats are able to keep fission products at higher temperatures than the steel cladding of the traditional fuel rods.

TABLE 8. PRESENTATIONS FROM SESSION 1.8. – INNOVATIVE REACTOR DESIGNS

Chair: **S. Monti** and **I. Tretyakov**

Id	Presenter	Country	Title
CN245-38 PPT-38	A. Sedov	Russian Federation	A Concept of VVER-SCP reactor with fast neutron spectrum and self-provision by secondary fuel
CN245-426 PPT-426	S. Bortot	Sweden	Design of a nitride-fuelled lead fast reactor for MA transmutation
CN245-531 PPT-531	K. Arie	Japan	Innovative TRU Burning Fast Reactor Cycle Using Uranium-free TRU Metal Fuel - Core Design Progress
CN245-431 PPT-431	J. Wallenius	Sweden	SEALER: a small lead cooled reactor for power production in the Canadian Arctic
CN245-303 PPT-303	N. Maslov	Russian Federation	Improving inherent safety BN-800 by the use of fuel assembly with (U, Pu)C microfuel.

5.1.9. Track 1. Poster Session

The summary of the papers in Track 1 presented in the Poster Session 1 is discussed in this section. There was a total of 18 papers (six from the Russian Federation, four from the Republic of Korea, two from Switzerland, one from Belgium, one from France, one from Italy, one from the Islamic Republic of Iran and two from the USA).

The papers covered a wide range of topics related to innovative fast reactor designs, such as: (i) design overview of the ASTRID reactor and the innovative design options, (ii) SFRs with heterogeneous fuel assembly configuration, (iii) advanced energy conversion system for SFRs (iv) design evaluation of SFR components such as fuel assembly, steam generator, decay heat exchangers, etc., (v) SFR technology for hydrogen production, (vi) studies related to Lead cooled fast reactors (LFRs), (vii) modular and transportable nuclear power plants employing sodium coolant and also the regulatory approach for fast reactors, (viii) design of core for incinerating TRU with advanced accident tolerant cladding, (ix) molten salt reactors, (x) subcritical molten salt reactors (MSRs), (xi) carrier salts for MSRs, (xii) boiling water cooled travelling wave reactors, and (xiii) breed-and-burn fast reactors.

Different types of fast reactor and other innovative reactor types were discussed in detail in these papers. A few highlights of some of the poster papers include: optimized proton energy is suggested to be applied for nuclear waste incineration using subcritical accelerator driven MSRs; SFR components (i.e. fuel subassemblies) are evaluated through structural analysis for their dynamic behaviour; the DHR heat exchanger is evaluated through high temperature creep-fatigue damage assessment; the steam generator is evaluated through structural integrity assessment; the proposal on heterogeneous fuel assemblies consisting of fuel elements with MOX fuel and fuel elements with metallic uranium of natural composition or U–Zr alloy, which has the potential to become an intermediate variant during conversion to metal fuel core; the lead cooled SMR with an effective design to withstand challenging conditions of combined loss of flow and loss of heat sink resulting in extended grace periods for initiating countermeasures; salt compositions based on the alkaline fluorides for the MSR option, including discussion of solubility and physicochemical characteristics and the possibility to address the required parameters in a high temperature fast reactor for production of adequate quantities of hydrogen on the basis of thermochemical cycles or for high temperature electrolysis, alternative power conversion cycle, etc.

5.2. TRACK 2 – FAST REACTOR OPERATION AND DECOMMISSIONING

5.2.1. Session 2.1. Commissioning and Operating Experience of Fast Reactors – I

Session 2.1. comprised of five papers and one invited talk on experience gained on fast reactors operation and development of sensors for sodium application.

Y. Nosov (Russian Federation) presented an overview of the operating experience gained with BN-600 reactor in the Russian Federation. The author reported satisfactory performance of the reactor and the modifications carried out in the reactor based on experience gained in the incorporation of the sodium to air heat exchanger for decay heat removal. The commissioning strategy was also presented following the experiences during commissioning of the BN-800. The improvements made to the BN-800 on the basis of experience gained from operation of the BN-600 were also discussed.

F. Baqué (France) and **K. Aizawa (Japan)** presented the development works carried out for an under sodium scanner, with particular importance to the sensitivity of the instrument. Mr Baqué also presented the scheme for deployment of the scanner in the ASTRID reactor and noted that the sensors have the capability to withstand sodium temperatures corresponding to the fuel handling state.

I. Petrov (Russian Federation) presented a paper on the experience gained in the manufacture, erection and commissioning of large-sized components in the BN-800 reactor. He also detailed the challenges faced and how on-site adjustments were made during erection and commissioning.

S. Raghupathy (India) presented a paper on testing and qualification of the trailing cable system developed for Prototype Fast Breeder Reactor (PFBR) at Kalpakkam. The trailing cable system carries power and signal cables from the reactor during rotation of the large and small rotating plugs.

K. V. Suresh Kumar (India) presented a paper on the works carried out for safety upgrading of the Fast Breeder Test Reactor (FBTR) at Kalpakkam. The major upgrading works were requirements initiated post-Fukushima and included seismic qualification and those criteria required for meeting the present standards. On the basis of the review, FBTR life was further extended by five years.

TABLE 1. PRESENTATIONS FROM SESSION 2.1. – COMMISSIONING AND OPERATING EXPERIENCE OF FAST REACTORS - I

Chair: **K. V. Suresh Kumar** and **A. Filin**

Id	Presenter	Country	Title
CN245-553 PPT-553	Y. Nosov (Invited)	Russian Federation	USSR and Russian fast reactor operation through the example of the BN-600 reactor operating experience and peculiarities of the new generation BN-800 reactor power unit commissioning
CN245-417 PPT-417	F. Baque	France	Main R&D objectives and results for under-sodium inspection carriers – Example of the ASTRID matting exceptional inspection carrier.
CN245-267 PPT-267	K. Aizawa	Japan	Development of under sodium viewer for next generation sodium cooled fast reactor
CN245-425 PPT-425	I. Petrov	Russian Federation	Manufacture, Installation and Adjustment of the BN-800 Reactor Plant Equipment
CN245-323 PPT-323	S. Raghupathy	India	Testing and Qualification of Trailing Cable system for Prototype Fast Breeder Reactor
CN245-307 PPT-307	K. V. Suresh Kumar	India	Safety Upgradation of Fast Breeder Test Reactor

5.2.2. Session 2.2. Commissioning and Operating Experience of Fast Reactors – II

In Session 2.2., the link was made between operating experience gained with Sodium cooled Fast Reactors (SFRs) and the design principles for the conception of the next generation of SFRs.

K. V. Suresh Kumar (India) presented an overview of more than 30 years of operating experience of the Fast Breeder Test Reactor (FBTR) in Kalpakkam. This feedback experience presentation described commissioning and successful operation and various problems encountered in different areas during the last few years (positive reactivity transients, dropping of orifices from steam generators, detection and management of sodium–water reaction in the steam generator, etc.).

D. Lukyanov (Russian Federation) summarized the valuable experience accumulated during BN-600 operation and from BN-800 commissioning on the sectoral monitoring tightness system of fuel element claddings (SSKGO). This detection system includes the supports with ionization fission chambers and the use of modern measurement and computing facilities. The SSKGO showed high reliability and its use is envisaged for next generation of reactors (BN-1200, MBIR).

F. Baqué (France) showed how in-service inspection and repair of the ASTRID SFR requires a large R&D effort for selecting, developing and qualifying ultrasonic techniques and tools.

V. Arasappan (India) detailed the design enhancements in the architecture of computer-based systems, computer hardware, human-machine interface and sensors in the design of future SFRs in India (PFBR and future projects).

S. Takaya (Japan) proposed an original approach to basic principles of maintenance for prototype SFRs. On the basis of JAEA's experience gained with the Monju SFR prototype, Mr Takaya showed how to identify risks specific to the reactor type (SFR) and the prototype status, and noted that the maintenance grade of systems should be determined with due consideration of the risks through the application of a graded approach.

TABLE 2. PRESENTATIONS FROM SESSION 2.2. – COMMISSIONING AND OPERATING EXPERIENCE OF FAST REACTORS - II

Chair: **D. Settimo** and **A. Gulevich**

Id	Presenter	Country	Title
CN245-278 PPT-278	K. V. Suresh Kumar (Invited)	India	Operating experience of FBTR
CN245-279 PPT-279	F. Baque	France	R&D status on in-sodium ultrasonic transducers for ASTRID inspection
CN245-167 PPT-167	S. Takaya	Japan	Proposal of Basic Principles of Maintenance Management for Prototype Reactors
CN245-186 PPT-186	D. Lukyanov	Russian Federation	Experience of commissioning of the sectoral monitoring tightness system of fuel elements claddings (SSKGO) of RF BN-600, RF BN-800
CN245-318 PPT-318	V. Arasappan	India	Design modifications of Instrumentation & Control System of future FBRs

5.2.3. Session 2.3. Decommissioning of Fast Reactors and Waste Management

Presentations in Session 2.3. included four contributions.

D. Settimo (France) presented on Superphenix Dismantling: Lessons Learned.

P. Filliatre (France) discussed on Dependability of the Fission Chambers for the Neutron Flux Monitoring System of the French Gen IV SFR.

S. Belov (Russian Federation) presented the Arrangement of the BN-600 Reactor Core Refuelling at Transition to the Increased Fuel Burnup.

K. Butov (Russian Federation) presented the Industrial Exploitation of Testing Ground for Treatment of Radwaste of Alkaline Coolants under Decommissioning of Fast Research Reactors.

These four papers discussed French and Russian approaches to the decommissioning of fast commercial and research reactors.

The dismantling of a sodium cooled fast reactor presents specificities related to the presence of sodium and to the need to eliminate the coolant before undertaking the usual dismantling procedures. The procedures used to eliminate sodium in the Superphenix reactor primary vessel have been explained, including the use of robots to drill and empty some areas of the primary vessel where sodium was accumulated. This experience enables recommendations to be made in terms of future commercial reactor design, aiming at making their future dismantling easier.

As regards to decommissioning of the research reactors, the Russian programme for decommissioning of the BR-10 fast reactor was presented. According to this programme, a special installation was built for conversion of alkaline coolants used in this reactor into solid radioactive waste. Now, the conditioning of radwaste from secondary sodium and preparing the treatment of radwaste from primary sodium are being carried out. The last presentation discussed the removal of the residual radwaste alkaline coolant from the inside surfaces of individual pieces of the equipment.

TABLE 3. PRESENTATIONS FROM SESSION 2.3. – DECOMMISSIONING OF FAST REACTORS AND WASTE MANAGEMENT

Chair: **H. Ohshima** and **V. Bezzubtsev**

Id	Presenter	Country	Title
CN245-560 PPT-560	D. Settimo (Invited)	France	Superphenix dismantling - Status and lessons learned
CN245-101 PPT-101	P. Filliatre	France	Dependability of the fission chambers for the neutron flux monitoring system of the French Gen IV SFR
CN245-386 PPT-386	S. Belov	Russian Federation	Arrangement of the BN-600 reactor core refuelling at transition to the increased fuel burnup
CN245-456 PPT-456	K. Butov	Russian Federation	Industrial Exploitation of Testing Ground for Treatment of Radwaste of Alkaline Coolants under Decommissioning of Fast Research Reactors

5.2.4. Track 2. Poster Session

During the poster session for Track 2, details of some innovative techniques and materials were presented on the subject of inspection, monitoring and detection in support of the SFR operation. The topics included fuel cladding damage, core diagnostics, sodium leakage detectors, hydrogen detectors on argon cover gas, in-service inspection, etc. Other topics presented in Track 2 posters included details of the experience gained during construction and commissioning of the recent SFRs, namely, the PFBR (India), the CEFR (China, with the involvement of the Russian Federation entities), BN-type reactors (Russian Federation) and the experience and organizational insights gained during the design of new projects for future SFRs (ASTRID, France).

5.3. TRACK 3 – FAST REACTOR SAFETY

5.3.1. Session 3.1. Innovative Reactor Designs

This session comprised of six presentations, two from Japan, two from the Russian Federation and two from France.

K. Morita (Japan) described the status of safety research activity in the field of sodium-cooled fast reactors (SFRs) in Japan, mainly on severe accident related issues. The author explained that the core damage sequences are analyzed by applying probabilistic risk assessment methodology and categorized into typical accident phases, i.e., initiating phase, transitions phase, and material relocation and cooling phase. In order to utilize superior characteristics of sodium as coolant, achievement of in-vessel retention is one of important objective of safety design and evaluation for SFRs.

A. Vasile (France) presented a large range of activities developed both on experiments and modelling by China, France, Japan, Republic of Korea, the Russian Federation and Euratom in the framework of the GIF SFR Safety and Operation project. These include Chinese safety code for near nominal conditions which is validated with CEFR and other software for Gen IV reactors, CEA approach based on SIMMER, JAEA on PRA for decay heat and sodium concrete interactions, etc. Code extension at KAERI for metal fuelled SFR was reviewed in detail. Decay heat removal (DHR) systems through primary vessel were also discussed. IPPE experiments on the PLUTON facility for fuel pin failure under ULOF conditions were examined.

S. Beils (France) presented ASTRID project and the safety design guide currently applied for the choices of the design options. The goal of this approach and the selected provisions is to enhance the application of the safety. Some of the highlighted principles are the capability of the primary circuit to withstand a mechanical energy release, evaluation of the mobile radioactive content, potential release way analysis, nitrogen between main and safety vessels, isolation of the retention room inside the confinement. Enhanced retention capabilities via retention room at the earlier design phase were discussed. The simulation approach in order to evaluate the radiological consequences of a severe accident and the main hypothesis were presented and discussed.

A. Anfimov (Russian Federation) presented the safety assessment of BN-1200 by considering three types of beyond design accidents. The first is the loss of power (pumps in the primary and secondary circuits stop and no feed water supply), the second is reactivity introduction by withdrawing of two control rods and the third is the fuel assembly blockage accident simulated by SOKRAT-BN and EVKLID codes. Thanks to two passive shutdown systems (including hydraulically suspended rods), no protection measures on the population are required as released radioactivity are much smaller than limits. Exposure doses on the population were found considerably less than the regulated value of 5 mSv/person per year.

F. Payot (France) presented the feasibility study on future SAIGA (Severe Accident In-pile experiments for Gen-IV reactors and the ASTRID prototype) programme carried jointly by CEA and National Nuclear Centre of the Republic of Kazakhstan (NNC-RK). The study was focused on preparation of tests in IGC reactor to investigate the degradation of one or more fuel pins during Total Instantaneous Blockage (TIB) sequences in a fuel assembly and power sequences as in SCARABEE and CABRI with homogeneous pins. Discussions are in progress between CEA and NNC-RK to define three in-pile SAIGA tests in conditions close to those expected in ASTRID.

S. Kubo (Japan) described safety design concept for future large SFRs with oxide fuel. This presentation discussed SFRs development and safety-related issues after the Fukushima Dai-ichi accident. In the light of the lessons learned from the Fukushima Dai-ichi accident, it is essential to make efforts to improve safety by incorporating comprehensive, effective, and rational severe accident measures into the design, although safety evaluations for beyond design basis core damage accidents were conducted before the accident. Development of targets of GIF and FaCT project were taken into account. The new regulatory requirements in Japan and Safety Design Criteria (SDC) and Safety Design Guidelines (SDG) for Gen IV SFRs were highlighted.

TABLE 1. PRESENTATIONS FROM SESSION 3.1. – INNOVATIVE REACTOR DESIGNS

Chair: **P. Alekseev** and **A. Rineiski**

Id	Presenter	Country	Title
CN245-28 PPT-28	K. Morita (Invited)	Japan	The Status of Safety Research in the Field of Sodium cooled Fast Reactors in Japan
CN245-133 PPT-133	A. Vasile	France	Recent activities of the safety and operation project of the sodium cooled fast reactor in the Gen IV International Forum
CN245-476 PPT-476	S. Beils	France	ASTRID safety design: Radiological confinement improvements compared to previous SFRs
CN245-385 PPT-385	A. Anfimov	Russian Federation	Safety Assurance for BN-1200 Power Unit During Accidents
CN245-67 PPT-67	F. Payot	France	The SAIGA experimental programme to support the ASTRID Core Assessment in Severe Accident Conditions
CN245-164 PPT-164	S. Kubo	Japan	Study on Safety Design Concept for future Sodium cooled Fast Reactors in Japan

5.3.2. Session 3.2. Core Disruptive Accident

In this session, six papers on CDA studies were presented.

F. Bertrand (France) of the CEA discussed on the status of severe accident studies for the French demonstration SFR, ASTRID, with the main insights of the studies discussed in terms of mitigation strategy and mitigation device design.

S. Raghupathy (India) of IGCAR presented the IAEA coordinated research project on the source term study, highlighting problem definition and the approach used to estimate radioactivity release under severe accident conditions of SFRs.

S. Kang (Republic of Korea) of KAERI reported on the advanced development of metallic fuel models for a reactor safety analysis code, SAS4A, who emphasized some new capabilities of SAS4A, with results of severe accident analysis for the Korean Prototype SFR (PGSFR).

K. Velusamy (India) of IGCAR presented an analytical investigation on flow blockage in SFR subassemblies. He also showed the thermal hydraulic analysis of core material relocation to the grid plate, the inner vessel and the core catchers after a whole core accident.

M. Flad (Germany) of KIT presented a paper detailing a study on the post-disassembly expansion (PDE) phase of CDAs in an SFR with a discussion of the main PDE phenomena and event paths enhancing or mitigating the mechanical work potential.

K. Lee (Republic of Korea) of KAERI reported an assessment of transient over-power (TOP) accident in the PGSFR showing that the preliminary design of the PGSFR meets safety acceptance criteria with a sufficient margin during the TOP event.

TABLE 2. PRESENTATIONS FROM SESSION 3.2. – CORE DISRUPTIVE ACCIDENT

Chair: **K. Morita** and **A. Volkov**

Id	Presenter	Country	Title
CN245-410 PPT-410	F. Bertrand (Invited)	France	Status of severe accident studies at the end of the conceptual design: feedback on mitigation features
CN245-335 PPT-335	S. Raghupathy	India	Source Term Estimation for Radioactivity Release under Severe Accident Scenarios in Sodium cooled Fast Reactors
CN245-56 PPT-56	S. Kang	Korea, Republic of	Advances in the Development of the SAS4A Code Metallic Fuel Models for the Analysis of PGSFR Postulated Severe Accidents
CN245-320 PPT-320	K.Velusamy (Invited)	India	Computational modelling of flow blockage in fuel subassemblies and molten material relocation in sodium cooled fast reactors
CN245-483 PPT-483	M. Flad	Germany	Quantitative Evaluation of the Post Disassembly Energetics of a Hypothetical Core Disruptive Accident in a Sodium Cooled Fast Reactor
CN245-172 PPT-172	K. Lee	Korea, Republic of	An assessment of transient over-power accident in the PGSFR

5.3.3. Session 3.3. Probabilistic Safety Assessment

There were three presentations in Session 3.3.

P. Pillai (India) presented a proposal for dynamic reliability analysis methodology by using Monte-Carlo sampling. The cost of the sampling was discussed.

V. Rychkov (France) presented an application for dynamic PRA on sodium cooled fast reactor (development of PyCATSHOO code). A data applicability of the present meta-thermohydraulics model and applicability to a wide spectrum of event sequences were discussed, as well as a computational cost of the code. Furthermore, a quantification of model uncertainty that comes from the meta-model was also deliberated.

P. Antipin (Russian Federation) presented a comparison of level 1 PRA for BN-600, BN-800 and BN-1200 reactors. The paper shows a comparison of core damage frequency under an internal initiating event and internal and external hazards. Subsequent discussions addressed the detailed analysis of level 2 PRA for the BN series reactors with particular reference to the safety aspects of the reactors.

TABLE 3. PRESENTATIONS FROM SESSION 3.3. – PROBABILISTIC SAFETY ASSESSMENT

Chair: **P. Antipin** and **T. Takata**

Id	Presenter	Country	Title
CN245-189 PPT-189	P. Pillai	India	Development of Smart Component Based Framework for Dynamic Reliability Analysis of Nuclear Safety Systems
CN245-42 PPT-42	V. Rychkov	France	Dynamic probabilistic risk assessment at a design stage for a sodium fast reactor.
CN245-419 PPT-419	P. Antipin	Russian Federation	Probabilistic safety analysis results for BN reactor power units

5.3.4. Session 3.4. Sodium Leak/Fire and other Safety Issues

There were five presentations in Session 3.4.

A. Vinogradov (Russian Federation) presented a numerical–experimental research for justification of sodium fire safety of SFR.

M. Aoyagi (Japan) discussed the measures taken to prevent sodium fire accidents.

L. Lebel (France) discussed on past experiences and tests related to sodium fires. A lively question and answer session followed in relation to sodium fire form and spread, fire aerosol modelling, sodium fire analysis codes and related future R&D, fluid structure interaction and modelling assumptions used in the analysis. Most of the questions referred to specific technologies, which indicated the common understanding of the technology among the delegates.

M. Jeltsov (Sweden) discussed the analysis of Lead cooled Fast Reactor (LFR) related to the pool behaviour under seismic conditions. The CFD analysis of a potential phenomenon on the liquid metal free surface was presented.

K. Yoshimura (Japan) presented an analysis of specific event sequences related to a sodium leak.

TABLE 4. PRESENTATIONS FROM SESSION 3.4. – SODIUM LEAK/FIRE AND OTHER SAFETY ISSUES

Chair: **Y. Okano** and **Y. Shvetsov**

Id	Presenter	Country	Title
CN245-102 PPT-102	A. Vinogradov	Russian Federation	Numerical – experimental research in justification of fire (sodium) safety of sodium cooled fast reactors
CN245-93 PPT-93	M. Aoyagi	Japan	Identification of important phenomena under sodium fire accidents based on PIRT process
CN245-326 PPT-326	L. Lebel	France	Learning from 1970 and 1980-Era Sodium Fire Experiments
CN245-355 PPT-355	M. Jeltsov	Sweden	Seismic sloshing effects in lead cooled fast reactors
CN245-290 PPT-290	K. Yoshimura	Japan	Evaluation of multiple primary coolant leakages accidents in Monju with consideration of passive safety features

5.3.5. Session 3.5. General Safety Approach

Five presentations were delivered and discussed in Session 3.5., three from the Russian Federation, one from France and one from EC/JRC.

L. Bolshov (Russian Federation) presented approaches to the safety ensuring and justification of the new generation of liquid metal cooled fast reactors with sodium and lead coolants being currently developed in the Russian Federation. The optimal design solutions, minimization of potential nuclear hazard, application of passive elements affecting reactivity and passive systems of residual heat removal, increased the reliability of main equipment and safety systems ensure the enhanced safety of these reactors. It was shown that power units with BN-1200 and BREST-OD-300 reactors designed within the framework of the FTP NETNG (Federal Target Programme “Nuclear Energy Technologies of the New Generation”) fully comply with the safety requirements for the Gen IV reactors. The main directions of improving the software and regulatory framework outlined in the Russian Federation were presented, and the high potential role of international cooperation in this field was noted.

K. Tucek (EC, Joint Research Centre) reported on the Lead cooled Fast Reactor (LFR) provisional System Steering Committee (SSC) of the Generation IV International Forum (GIF) that proposed a set of Safety Design Criteria (SDC) dedicated to LFRs. The objective of the LFR-SDC is to prescribe a set of reference criteria for the design of LFR systems, structures, and components with the aim of achieving the safety goals of the Gen IV reactor systems. A set of reference safety design criteria for LFRs is systematically and comprehensively laid out in the SDC to facilitate the development, safety assessment and licensing of LFRs, including BREST-OD-300, ALFRED, SSTAR, SVBR-100, CLEAR-I, and MYRRHA. Mr Tucek summarized results of the steps taken to draft the present set of LFR SDC and provided an outlook for further review and development activities, in particular towards individual sets of detailed Safety Design Guidelines (SDGs).

A. Stremin (Russian Federation) presented on deterministic safety analysis of lead cooled fast reactor BREST-OD-300. Results of the safety analysis of the reactor were presented for up to four OE Violations in Normal Operation (VNO). Selected VNO initiating events, accompanied by the greatest disturbance and deeper relative to the nominal power deviations of parameters are important to safety. As the security criteria of the reactor facility in violation of the normal operation, the exceedance of the established design limits of the power unit parameters were taken. Considered series of normal operation systems’ and safety systems’ failures consisted of a minimum of ten failures. The analysis showed a high level of safety of the unit with BREST-OD-300 reactor and its resistance to accidents with multiple probable simultaneous failures of the systems and components.

A. Bochkarev (Russian Federation) presented the inherent safety performance of 2800 MW(th) sodium cooled fast reactor with MOX core during anticipated transient without scram (ATWS) initiated by various accident initiating events with simultaneous failure of all shutdown systems in all cases under investigation. The impact of various safety features on SFR inherent safety performance during ATWS was also analyzed. The decrease in hydraulic resistance of primary loop, increase in primary pump coast-down, the implementing of thermo-mechanical, leakage based and other self-actuated safety systems considered as additional natural feedbacks were taken into account. Performing analysis resulted in a set of recommendations to the characteristics of the features referred above for the purpose of enhancing the inherent safety performance of SFR under investigation.

A. Saturnin (France) discussed evolution in estimations of the collective radiation dose for the nuclear reactors from the Gen II through to the Gen IV, based on publications from the NEA and the IAEA. The external individual doses received by the personnel were measured and recorded, in conformity with the regulations in force. The sum of these measurements enabled an evaluation of the annual collective dose expressed in man•Sv/year. This information is a useful tool when comparing the different design types and reactors. The spread of good practices (optimization of working conditions and of the organization, sharing of lessons learned, etc.) and ongoing improvements in reactor design meant that over time, the doses of various origins received by the personnel decreased. In the case of sodium cooled fast reactors, the compilation and summarizing of various documentary resources enabled them to be compared to other types of reactors of the second and third generations. From these results, it could be seen that the doses received during the operation of SFR are significantly lower for this type of reactor.

TABLE 5. PRESENTATIONS FROM SESSION 3.5. – GENERAL SAFETY APPROACH

Chair: **L. Bolshov** and **P. Gauthe**

Id	Presenter	Country	Title
CN245-577 PPT-577	L. Bolshov (Invited)	Russian Federation	Safety assurance of the new generation of the Russian fast liquid metal reactors
CN245-75 PPT-75	K. Tucek	EC/JRC	Development of Safety Design Criteria for the Lead cooled Fast Reactor
CN245-549 PPT-549	A. Stremin	Russian Federation	Deterministic safety analysis of reactor BREST-OD-300
CN245-463 PPT-463	A. Bochkarev	Russian Federation	SFR inherent safety features and criteria analysis
CN245-16 PPT-16	A. Saturnin	France	Evolution of the collective radiation dose from the nuclear reactors through the 2nd to the 4th generation.

5.3.6. Session 3.6. Safety Analysis

Session 3.6. comprised six presentations on a wide variety of subjects made by experts representing France, India, Japan, the Russian Federation and Sweden.

A. Yamaguchi (Japan) from the University of Tokyo reported very thoroughly on a large number of ongoing projects, both experimental and computational, that the sodium reactor thermohydraulics research team in Japan are leading. Following the presentation, there were discussions and questions specifically regarding the work on bubble formation in water sodium mixture systems.

C. Sowrinathan (India) from IGCAR delivered a presentation on the safety aspects and limitations on the use of metallic fuel in sodium cooled fast reactors. The details were given of the Indian approach to determining the operational temperature and stress limits and methods on how to calculate cladding thinning during standard operation and under transient conditions. The IGCAR estimate of the likely frequency and duration of different types of transient condition were also discussed.

R. Chalyy (Russian Federation) from IBRAE, presented the capabilities and applications of the SOCRAT-BN code suite for fast reactor safety analysis. After presenting the types of analysis possible with SOCRAT-BN and the various methods employed, a study of an extremely severe postulated transient for the proposed BN-1200 reactor was presented. This transient involved the rapid introduction of a very large amount of positive reactivity due to the withdrawal of control rod banks. SOCRAT-BN was able to model each stage of the severe accident, which would lead to core melt and limited release of radioactivity (although below regulatory margins). Intense discussions followed on the BN-1200 simulation using the SOCAR-BN code.

A. Vasile (France) from CEA presented a safety analysis of the ALLEGRO fast gas cooled reactor system using the CATHARE code. Mr Vasile showed that the present design (75 MW(th)) could not reach the desired safety performance in all postulated transient conditions, and that core designs of smaller power output (26 MW(th)) would have superior safety performance. The ensuing discussion focused on to what extent it would be possible to redesign (or upgrade) the decay heat removal system rather than redesigning the core itself.

S. Qvist (Sweden), presented a research project led by the University of California Berkeley and Argonne National Laboratory supported by Uppsala University (Sweden) and introduced a new passive safety system for fast reactors. The author presented the design and function of Autonomous Reactivity Control (ARC) systems, as well a range of transient analysis results of ARC equipped cores using the SASSYS/SAS4A and CHD transient analysis code packages. The results showed that with ARC systems, adequate passive safety performance may be achieved for the postulated unprotected transients in fast reactors. The discussion focused on how the ARC systems could be tested in-situ and whether there could be a risk of gas leakage from the system.

P. Gauthe (France) from CEA presented a simple but powerful analytical approach for estimating the safety performance of fast reactors. Using a set of equations, the asymptotic state of the core can be analyzed in various transients, and the guiding principles for safety performance can be identified by ratios and inequalities developed by the author. The discussion focused on a comparison of the author’s methods with those of the quasi-static reactivity balance methods developed in the USA. Mr Gauthe explained that although the approaches are similar, the main difference is in the treatment of the primary coolant flow rate.

TABLE 6. PRESENTATIONS FROM SESSION 3.6. – SAFETY ANALYSIS

Chair: **S. Qvist** and **Y. Khomyakov**

Id	Presenter	Country	Title
CN245-354 PPT-354	A. Yamaguchi (Invited)	Japan	Current Thermal Hydraulic Activities on Sodium cooled Fast Reactors in Japan
CN245-364 PPT-364	C. Sowrinathan	India	Design Safety Limits for Transients in a Metal Fuelled Reactor
CN245-281 PPT-281	R. Chalyy	Russian Federation	SOCRAT-BN integral code for safety analysis of NPP with sodium cooled fast reactors: development and plant applications
CN245-64 PPT-64	A. Vasile	France	Thermal-hydraulics and Decay Heat Removal in GFR ALLEGRO
CN245-557 PPT-557	S. Qvist	Sweden	Autonomous Reactivity Control
CN245-123 PPT-123	P. Gauthe	France	Sensitivity studies of SFR unprotected transients with global neutronic feedback coefficients

5.3.7. Session 3.7. Core Disruptive Accident Prevention

Session 3.7. comprised four presentations focused on the technical advances needed to prevent accident propagation to the severe condition by applying basic principles and state of the art design technologies.

C. Kim (Republic of Korea) presented passive safety devices such as Static Absorber Feedback Equipment (SAFE) and Floating Absorber for Safety at Transient (FAST). These innovative concepts permit enhancement of the safety of an SFR by lowering the core void reactivity and by providing additional negative reactivity into the core.

L. Costes (France) summarized the progress on the prevention of a severe accident in the ASTRID design that was achieved primarily through the provision of a highly reliable reactivity control function by two diverse shutdown systems with related design criteria giving sufficient safety margins for different plant states. Furthermore, the inherent behaviour of reactors to cope with hypothetical transients is implemented by involving special devices acting in the case of loss of flow or core heating during severe accidents. The loss of decay heat removal function is also practically eliminated in order to prevent a severe accident by diversification of the DHRS.

M. Sarotto (Italy) presented the global and local effects due to the accidental withdrawal of one control rod in the ALFRED core analyzed by evaluating the impact in terms of reactivity balance and power distribution. Detailed temperature distributions in all the pins and surrounding subchannels were obtained through thermohydraulic studies with the RELAP5 system code and the ANTEO+ subchannel code combined with a three-dimensional power map from the ERANOS code. These results verified the compliance of ALFRED within the safety limits of both fuel and clad, even in a completely unprotected scenario.

Y. Khomyakov (Russian Federation) discussed on the reduced maximum reactivity margin as an important design target necessary to rule out an accident that leads to potential population evacuation. In the PRORYV project, the analysis of two specific designs — BN-1200 SFR and BREST-OD-300 LFR — demonstrated the advantages of the low reactivity margin, which results in a reduced possibility of fuel melting in those cases involving inadvertent insertion of all the reactivity margin without reactor scram.

TABLE 7. PRESENTATIONS FROM SESSION 3.7. – CORE DISRUPTIVE ACCIDENT PREVENTION

Chair: **H. Y. Jeong** and **Y. Ashurko**

Id	Presenter	Country	Title
CN245-284 PPT-284	C. Kim (Invited)	Korea, Republic of	Optimization of Passive Safety Devices FAST and SAFE for Sodium cooled Fast Reactors
CN245-474 PPT-474	L. Costes	France	ASTRID safety design: Progress on prevention of severe accident
CN245-182 PPT-182	M. Sarotto	Italy	Impact of an accidental control rod withdrawal on the ALFRED core: tridimensional neutronic and thermal-hydraulic analyses
CN245-131 PPT-131	Y. Khomyakov (Invited)	Russian Federation	Minimisation of Reactivity Margin for Equilibrium Core of Liquid Metal Cooled Fast Reactors

5.3.8. Track 3. Poster Session

In Track 3, a total of 33 posters were presented on fast reactor safety. The papers were on a wide range of topics related to fast reactor safety, including but not limited to safety by design approaches, safety of equipment, harmonization of safety requirements and approaches, probabilistic and deterministic approaches and studies, severe accidents analysis and simulation. A few highlights of some of the poster papers included: Model validation of the ASTERIA-FBR code related to core expansion phase based on THINA experimental results; passive complementary safety devices for ASTRID severe accident prevention; decay heat removal system in the secondary circuit of the sodium cooled fast reactor and evaluation of its capacity; design and development of stroke limiting device (SLD) for control & safety rod drive mechanisms (CSRDMs) of future FBRs; thermal hydraulic investigation of sodium fire and hydrogen production in top shield enclosure of an FBR following a core disruptive accident; preliminary safety performance assessment of ESFR CONF-2 sphere-pac-fueled core; probabilistic safety analysis of NPP with BREST-OD-300 reactor; chugging boiling in low-void SFR core; and new phenomenology of unprotected loss of flow transient.

5.4. TRACK 4 – FUEL CYCLE SUSTAINABILITY, ENVIRONMENTAL CONSIDERATIONS AND WASTE MANAGEMENT

5.4.1. Session 4.1. Fuel Cycle Overview

Session 4.1. focused on an overview of fuel cycles and comprised of six technical presentations originating from four countries (France, India, the Russian Federation and Switzerland) and one international organization (OECD/Nuclear Energy Agency).

K. Jayaraman (India) and **A. Shardin (Russian Federation)** covered an overview of some recycling technology developments and achievements.

C. Poinssot (France) covered the specific case of minor actinides, in France.

These presentations illustrated the significance of these R&D programmes for developing efficient and innovative recycling processes for dealing with high burnup, high plutonium contents, fast neutron reactor fuels (oxides but also nitrides or carbides). Developments are focused on disassembly, on the head-end processes (dissolution) and on the separation steps which remain, in most cases, based on hydro processes, although the Russian Federation is proposing a combination of pyro and hydro technologies. France also presented an overview of its achievements in the field of minor actinide separation after 25 years of research and the motivation for refocusing its R&D efforts towards plutonium multi-recycling with limited involvement.

J. Krepel (Switzerland) and **C. Poinssot (France)** focused on the development of a systemic approach for benchmarking different types of nuclear energy systems and their associated fuel cycles, either from a general viewpoint (overall efficiency, safety, etc.) or on the specific issues of their respective environmental impacts. These two presentations evidenced the overall potential benefit of Gen IV nuclear energy systems, in particular for reducing the environmental impact.

S. Cornet (OECD/NEA) delivered an overview of the collaborative works that NEA is supporting in the field of advanced fuel cycles through its working groups and the technical documents that these groups are preparing.

The session was concluded by a question and answer discussion.

TABLE 1. PRESENTATIONS FROM SESSION 4.1. – FUEL CYCLE OVERVIEW

Chair: **C. Poinssot** and **V. Vidanov**

Id	Presenter	Country	Title
CN245-507 PPT-507	C. Poinssot (Invited)	France	1992-2017: 25 years of success story for the Development of Minor Actinides Partitioning Processes
CN245-52 PPT-52	J. Krepel	Switzerland	Comparison of fast reactors performance in the closed U-Pu and Th-U cycle
CN245-76 PPT-76	A. Shadrin (Invited)	Russian Federation	Reprocessing of fast reactors mixed U-Pu used nuclear fuel: studies and industrial test
CN245-527 PPT-527	S. Cornet	NEA	Overview of the Nuclear Energy Agency Scientific Activities on Advanced Fuel Cycles
CN245-506 PPT-506	C. Poinssot	France	Assessment of the anticipated improvement of the environmental footprint of future nuclear energy systems
CN245-297 PPT-297	K. Jayaraman	India	Concurrent Trends in Indian Fast Reactor Fuel Reprocessing Programme

5.4.2. Session 4.2. Reprocessing and Partitioning

Session 4.2. comprised four presentations, three from France and one from the Russian Federation.

C. Venard (France) presented the capabilities of the ASTRID reactor to use plutonium derived from PWR MOX fuel reprocessing. The innovative CFV core (low sodium void coefficient) of ASTRID allowed the use of reprocessed PWR MOX fuels. The presentation focused on the CFV BD 16/10 core, its performance and the characteristics of fresh and spent fuel containing plutonium from PWR MOX fuels.

E. Buravand (France) presented a method for the dissolution of fast reactor MOX spent fuel in order to recover plutonium. A method of dissolution with Ag (II) was presented. The digestion step allows the recovery of 99% of residual Pu.

A. Shadrin (Russian Federation) reported that a closed nuclear fuel cycle is currently being studied for the BREST-300 fast reactor using nitride fuel. A combined pyro + hydro technology known as the PH-process is being developed for recycling nitride fuel. The different methods of denitration investigated in the project were presented.

N. Reynier-Tronche (France) presented the results of a dissolution study carried out on spent SFR MOX fuel (NESTOR-3 pin). Dissolution experiments were carried out on irradiated materials. The masses of dissolution residues were measured, and the yield of dissolved and undissolved plutonium was calculated. Results showed that plutonium can be dissolved in high yield. The method showed that the more irradiated the fuel is, the more easily plutonium can be dissolved, but also the higher the residue mass is, owing to the greater quantity of fission products.

TABLE 2. PRESENTATIONS FROM SESSION 4.2. – REPROCESSING AND PARTITIONING

Chair: **S. Cornet** and **A. Glazov**

Id	Presenter	Country	Title
CN245-348 PPT-348	C. Venard	France	Pu recycling capabilities of ASTRID reactor
CN245-520 PPT-520	E. Buravand	France	First assessment of a digestion method applied to recover plutonium from refractory residues after dissolving spent SFR MOX fuel in nitric acid
CN245-114 PPT-114	A. Shadrin	Russian Federation	The actinide oxides preparation by thermal denitration
CN245-519 PPT-519	N. Reynier-Tronche	France	A comprehensive study of the dissolution of spent SFR MOX fuel in boiling nitric acid (the PHENIX NESTOR-3 case)

5.4.3. Session 4.3. Partitioning and Sustainability

Session 4.3. comprised five presentations, three from the Russian Federation, one from Hungary and one from India.

K. A. Venkatesan (India) presented an overview of the R&D activities carried out at the Indira Gandhi Centre for Atomic Research (IGCAR) towards the development of advanced flow-sheets for trivalent actinide group separation, lanthanide-actinide separation and demonstrations with real HAW from FR using TRUEX solvent formulation. Decontamination of ruthenium from the solvent was achieved through a cleanup stage with sodium carbonate and anion exchange resin before recycling. A new unsymmetrical DGA (D^3DODGA) in n-dodecane showed the same excellent extraction properties than TODGA using simulated HAW from FR, without the addition of any modifier to avoid third phase formation during the extraction process. The extraction of trivalent transition metals is avoided by complexing them in the aqueous phase. All extracted trivalent actinides and lanthanides are easily stripped by using dilute nitric acid.

A. M. Potapov (Russian Federation) presented scientific and technological aspects of pyrochemical recycling of nitride SNF from the fast reactor in molten salts. Experimental results of the chlorination of uranium nitride pellets with cadmium shown that the increase of the temperature of the bath has a positive effect on the efficiency of the chlorination process.

L. Tkachenko (Russian Federation) presented a dynamic test using mixer-settlers to separate actinides(III) from HAW issued from PUREX process using a solvent based on TODGA in meta-nitrobenzotrifluoride (F-3). About 99.97% Am(III) was recovered with high decontamination rates for the rare earth elements. Technetium was found accumulating in the recycle solvent, so further investigations are needed to improve the process.

M. Halász (Hungary) presented results obtained in developing burnup models for the Gen IV reactors (GFR, LFR and SFR) using FITXS scheme developed at the BME Institute of Nuclear Techniques. The model was integrated into fast reactor fuel cycle model (SITON v2.0), developed at the HAS Center for Energy Research, and the closed cycle equilibrium parameters of the reactors were investigated for a GFR. The analyzed EPR-GFR2400 transition fuel cycle shows that the GFR2400 needed plutonium from the spent EPR fuel to start and operate and that it can consume all MAs originated from the spent EPR fuel, however, the MA reduction is limited by MAs left in the last discharged batches.

V. Vidanov (Russian Federation) presented the americium and curium separation from HAC produced by reprocessing SNF from WWER-440, using two-stages technology, based on extraction chromatography. The first stage consists of separating and concentrating Am-Cm fraction using KY-2-8 resin and DTPA and HTA solution at pH 7-8 for elution. The second stage consists on separating americium from curiumusing “Tokem-308” resin and DTPA solution at pH 7,5 for elution. Pure fraction of americium was obtained with a content of corium lower than 0.8% by mass and a content of ^{154,155}Eu lower than 0.1% by activity.

E. Lyman (USA) presented the concerns of the Union of Concerned Scientists (UCS) about the readiness of pyroprocessing technologies to be ready for commercial development based on information published in the USA reports.

TABLE 3. PRESENTATIONS FROM SESSION 4.3. – PARTITIONING AND SUSTAINABILITY

Chair: **A. Gonzalez-Espartero** and **A. Shadrin**

Id	Presenter	Country	Title
CN245-315 PPT-315	K. A. Venkatesan	India	Advanced flow-sheet for partitioning of trivalent actinides from fast reactor high active waste
CN245-259 PPT-259	A. Potapov	Russian Federation	Pyrochemical recycling of the nitride SNF of fast neutron reactors in molten salts as a part of the short-circuited nuclear fuel cycle
CN245-228 PPT-228	L. Tkachenko	Russian Federation	Dynamic test of extraction process for americium partitioning from the PUREX raffinate
CN245-111 PPT-111	M. Halász	Hungary	Fuel cycle studies of Generation IV fast reactors with the SITON v2.0 code and the FITXS burnup scheme
CN245-237 PPT-237	V. Vidanov	Russian Federation	Hot test of technique separation of americium and curium
CN245-492 PPT-492	E. Lyman	USA	External Assessment of the U.S. Sodium-Bonded Spent Fuel Treatment Program

5.4.4. Track 4. Poster Session

Track 4 included fourteen posters from the Russian Federation and one poster each from India, Japan and Poland. Of the seventeen posters, four were on mathematical modelling, three on environmental aspects, two on fuel cycle strategy/reprocessing, two on hydrometallurgy, two on pyrochemistry and one on partitioning and transmutation. Four posters dealing with modelling suggested a move away from conducting experimental exercises towards modelling.

The intensive programmes of study being undertaken were focused on:

- Developing efficient and innovative recycling processes for dealing with high burnup, high plutonium contents, fast neutron reactor fuels (oxides, nitrides or carbides) by hydro, pyro and a combination of pyro and hydro technologies;
- Minor actinide recovery and separation for transmutation; and
- Technologies for plutonium multi-recycling with limited involvement.

The principal efforts are directed at:

- Coherence decreasing of nuclear cycle influence on the environment; and
- Improving the economic efficiency of fuel reprocessing technologies.

Countries have different approaches for R&D in spent fuel reprocessing depending on the type of fuel selected over short and long-time perspectives. The poster session for Track 4 showed a trend from experimentation to modelling.

5.5. TRACK 5 – FAST REACTOR MATERIALS (FUELS AND STRUCTURES) AND TECHNOLOGY

5.5.1. Session 5.1. Advanced Fast Reactor Fuel Development – I

Session 5.1. comprised six presentations, two from Republic of Korea, two from the Russian Federation, one from France and one from India.

J. Park (Republic of Korea) presented a fabrication process of metallic fuels for Sodium cooled Fast Reactors (SFR) developed using the injection casting. U-Zr-RE(Nd-Ce-Pr-La) fuel slugs were fabricated and characterized to optimize the injection casting process. The microstructure examined by SEM showed that inclusions were uniformly distributed over the fuel slug. The reaction between the melt and the crucible was found to be significant in the fabrication of rare earths (RE)-containing fuel slugs compared to U-Zr fuel slugs. The pressurized injection casting method was also developed to fabricate the fuel slugs containing volatile elements. U-Zr-Mn fuel slugs were fabricated as a surrogate for Am-bearing metallic fuels under three different melting pressure conditions. From the chemical composition analysis by the ICP-AES method, no evaporation of manganese was detected in the fuel slugs fabricated under argon atmosphere higher than 400 torr.

B. Tasarov (Russian Federation) presented a new concept for metal fuel fabrication for fast reactors lowering swelling effect. To get that, it is proposed to create an open porosity of 15 to 25% of the entire volume of the fuel pellet by applying the technique of powder metallurgy based on electromagnetic compaction methods. The particle size from 15 μm to 3000 mm was made from alloys uranium with molybdenum and zirconium by mechanical means, and by means of a hydrogenation-dehydrogenation. It was shown that the optimum powder fabrication technology is the mechanical grinding followed by grinding in a ball mill. Pycnometric analysis showed the presence of open (connected) porosity with weak dependence of density on the pressing pressure, and the size of the starting powders. Measurement of thermal conductivity of porous fuel pellets showed the thermal conductivity is decreasing with increasing porosity. Thermophysical calculation of temperature fields in a fuel rod showed the possibility of using porous metal fuel in fast reactors.

L. Zabudko (Russian Federation) presented the development of uranium-plutonium nitride fuel. The work has been carried out in accordance with the comprehensive programme of computational and experimental validation of the performance of mixed nitride fuel for BN-1200 and BREST-OD-300. The technology of manufacturing pellets of mixed uranium-plutonium nitride fuel by carbothermic synthesis method had been developed. More than 500 fuel elements were produced to be tested in MIR, BOR-60 and BN-600 reactors (more than 60 fuel elements of different modifications with mixed nitride fuel were loaded into the BOR-60 and 15 experimental fuel assemblies with mixed nitride fuel were delivered to the BN-600). Researches are continuing to improve the composition and structure of MNUP fuel in order to increase ductility, reduce crack resistance and fuel swelling speed. An industrial production is scheduled to be launched by 2020. Post-irradiation examinations confirmed that all fuel elements remained at their work capacity.

V. Blanc (France) presented the design of the fuel and radial shielding sub-assemblies for the ASTRID CFV v4 core at the end of the conceptual design phase (AVP2). Innovative design choices have been made to meet the ASTRID project requirements, marking a break with the former Phenix and SuperPhenix French SFRs. Fuel sub-assemblies ensure a low sodium void worth (CFV core), thanks to axially heterogeneous fuel pins, a wide cladding/small spacer wire bundle, a sodium plenum above the fuel pins, and upper neutron shielding with B4C sodium-bonded pins. The upper neutron shielding helps to reach a low secondary sodium activity level and would be made removable on-line through the assembly head so as to meet washing constraints. Studies had been performed to increase the stiffness of the stamped spacer pads on the wrapper tube in order to analyze its effect on the core mechanical behaviour during hypothetical radial core flowering and compaction events. ASTRID specification for ^{24}Na activity in secondary loops appears to be reachable.

B. Nashine (India) presented the development of electromagnetic devices for sodium cooled fast reactors such as electromagnetic pumps, magnetic flow meters and sodium level probe. The design and development of sodium submersible annular linear induction pump (ALIP) by employing mineral insulated (MI) cable for fabricating winding of ALIP addressed the problem of development and non-availability of high temperature electromagnetic pump (ALIP) for draining of primary radioactive sodium from main vessel of pool type sodium cooled fast reactor. The submersible ALIP can be also used for pumping sodium in integrated cold trap system submerged in primary sodium of pool type sodium cooled fast reactor and any other application where pumping is needed in pump submerged condition. Use of samarium cobalt magnet in flowmeter has facilitated reduction in weight and increase in sensitivity. Similarly, the development of electromagnet based flowmeter has overcome constraints of high temperature operation. Development of eddy current based ex-vessel allows the measurement of discrete sodium level in side vessel without need of penetration in the vessel.

J. Kim (Republic of Korea) presented the chemical interaction of irradiated metallic fuel and T92 cladding conducted at 750°C. In the case of U-10Zr slug with T92 specimen, eutectic reaction layer was observed and element distribution indicates that significant migration of elements occurs and neodymium, plays a significant role in increasing penetration depth. The measured penetration rate is almost similar but slightly higher than the reference value. It is thought that the difference comes from the furnace cooling. On the other hand, no eutectic melting region was found in the case of irradiated U-10Zr-5Ce with T92 specimens by fuel slug oxidation. Therefore, to minimize the oxidation of specimen at high temperatures, special rig for the heating test was made. Preliminary examination using the rig showed that rig is effective in preventing the oxidation.

TABLE 1. PRESENTATIONS FROM SESSION 5.1. – ADVANCED FAST REACTOR FUEL DEVELOPMENT – I

Chair: **V. Troyanov** and **C. Sowrinathan**

Id	Presenter	Country	Title
CN245-198 PPT-198	J. Park (Invited)	Korea, Republic of	Fabrication Characteristics of Injection-cast Metallic Fuels
CN245-347 PPT-347	B. Tarasov	Russian Federation	Metal fuel for fast reactors, a new concept
CN245-62 PPT-62	L. Zabudko	Russian Federation	Development of innovative fast reactor nitride fuel in Russian Federation: state-of-the-art
CN245-128 PPT-128	V. Blanc	France	Conceptual design of fuel and radial shielding sub-assemblies for ASTRID
CN245-174 PPT-174	B. Nashine	India	Development of Electromagnetic Devices for Sodium Cooled Fast Reactor Application
CN245-106 PPT-106	J. Kim	Korea, Republic of	Fuel Cladding Chemical Interaction Tests of Irradiated Metallic Fuel

5.5.2. Session 5.2. Advanced Fast Reactor Fuel Development – II

Session 5.2. was devoted to consideration of recent progress in advanced fast reactor fuel development.

O. Azpitarte (Argentina) from the National Atomic Energy Commission (CNEA) discussed on the International Atomic Energy Agency (IAEA) Coordinated Research Project (CRP) on "Sodium Properties and Safe Operation of Experimental Facilities in Support of the Development and Deployment of Sodium cooled Fast Reactors - NAPRO". The CRP is covered by three work packages, focused on the compilation and expert assessment of data sets of sodium physical and chemical properties, as well as correlations for pressure drops and heat transfer in sodium facilities; the compilation, evaluation and development of best practices and guidelines for the design, operation and maintenance of sodium facilities, and finally on the compilation and development of guidelines and rules for the safe operation of sodium facilities, including, among others, the prevention, detection and mitigation of sodium leaks and fires.

B. Guillou (France) from ALSYMEX briefly described several realizations of the firm in the nuclear field. Mr Guillou followed with the presentation of the hot cell development for ASTRID which interfaces with various parts of the reactor operations chain. An innovative solution of a robotic arm for fuel elements dismantling was proposed.

V. Blanc (France) presented the framework of the basic design of the advanced sodium technological reactor for industrial demonstration (ASTRID) project, the design criterion for fuel melting margin during nominal operation conditions, given by the melting probability. Oxide fuel temperature and melting temperature calculated with the CEA fuel performance code GERMINAL, strongly depend on parameters from manufacturing processes, irradiation conditions and fuel. Uncertainties associated to these parameters in the melting margin evaluation and the sensitivity to these parameters were calculated using a statistical approach (the simulation platform URANIE).

K. Tucek (EC) from Joint Research Centre presented investigations in the evaluation of irradiated mixed oxide fuel properties. These evaluations, conducted in the framework of the ESNII+ program, were based on the NESTOR3 tests pins irradiated in Phenix. A good agreement was found with previous results, as example for the thermal conductivity and its variation with temperature and burnup. This project, ended in 2016, would continue under the ESFR-SMART project in order to enlarge the PIE examinations and analyze other properties like melting temperature.

K. V. Suresh Kumar (India) from Indira Gandhi Centre for Atomic Research introduced an extensive experience of India with mixed carbide fuels at the fast breeder test reactor PFBR at Kalpakkam that has been operating for over 25 years. Selected recent results of post-irradiation examinations (PIE) at various stages up to this high burnup on fission product migration, gas release, fuel swelling behaviour and microstructural evolution of the mixed-carbide fuel were discussed.

S. Porollo (Russian Federation) from the IPPE, was also focused on the post irradiation examination of fuel, in particular the analysis of nitride fuel swelling and the fission products behaviour. Results were based on the observation of 11 pins, 8 standards and 3 experimental pins irradiated in BR-10. These experimental results would be taken into account in order to validate the sensibility of swelling and fission gas release to temperature and initial fuel density in the DRAKON code.

TABLE 2. PRESENTATIONS FROM SESSION 5.2. – ADVANCED FAST REACTOR FUEL DEVELOPMENT – II

Chair: **M. Veshchunov and V. Blanc**

Id	Presenter	Country	Title
CN245-458 PPT-458	O. Azpitarte	Argentina	The IAEA Coordinated Research Project on Sodium Properties and Safe Operation of Experimental Facilities in Support of the Development and Deployment of Sodium cooled Fast Reactors (NAPRO)
CN245-407 PPT-407	B. Guillou	France	Preliminary Basic Design of ASTRID Hot Cells.
CN245-333 PPT-333	V. Blanc	France	Fuel Melting Margin Assessment of Fast Reactor Oxide Fuel Pins using a Statistical Approach
CN245-525 PPT-525	K. Tucek	EC	New catalogue on (U, Pu) O_2 properties for fast reactors and first measurements on irradiated and non-irradiated fuels within the ESNII+ project
CN245-252 PPT-252	K. V. Suresh Kumar	India	Fission product and swelling behaviour in FBTR mixed carbide fuel
CN245-81 PPT-81	S. Porollo	Russian Federation	Analysis of experimental data on fission gas release and swelling in mononitride fuel irradiated in BR-10 reactor

5.5.3. Session 5.3. Advanced Fast Reactor Cladding Development – I

Five presentations were made in the session by the IAEA, Japan (JAEA) and the Russian Federation. The presentations covered past achievements, current activities and future views of fast reactor structural materials. They included both experimental studies and analytical studies and indicate that significant effort has been made for advanced structural materials development in Japan and the Russian Federation.

T. Asayama (Japan) summarized ongoing efforts in the JAEA on the development of core and structural materials for sodium cooled fast reactors and also described the current status of codification of structural materials standards in the design code of fast breeder reactors published by the Japan Society of Mechanical Engineers.

V. Bobrovskii (Russian Federation) and **A. Kozlov (Russian Federation)** were devoted to the post-irradiation examination data of BN-600 reactor fuel pin claddings. The principal results of BN-600 structural materials examination was given as well as a review of the developed theoretical concepts.

A. Sorokin (Russian Federation) proposed modelling of fracture strain and fracture toughness of irradiated austenitic steels over a wide range of temperatures with particular regard to swelling and thermal ageing. The model was developed for prediction of both quasi-brittle intergranular and ductile transgranular fracture and for the fracture mechanism transition.

M. Veshchunov (IAEA) devoted his presentation to the IAEA activities in the area of nuclear power reactor fuels. The report provided information about organization and implementation practices of these activities and summarized their major outputs, including ongoing Coordinated Research Projects (CRPs) and technical meetings in the area of fuel engineering.

TABLE 3. PRESENTATIONS FROM SESSION 5.3. – ADVANCED FAST REACTOR CLADDING DEVELOPMENT – I

Chair: **L. Zabudko** and **J. Park**

Id	Presenter	Country	Title
CN245-77 PPT-77	T. Asayama (Invited)	Japan	Development of core and structural materials for fast reactors
CN245-32 PPT-32	V. Bobrovskii	Russian Federation	Results of monitoring, using high-resolution neutron diffraction, of radiation-induced damages in claddings of fuel pins after their performance in the reactor BN-600 as a ground for prolongation of their life expectancy
CN245-153 PPT-153	A. Sorokin	Russian Federation	Fracture strain and fracture toughness prediction for irradiated austenitic steels over wide range of temperatures taking into account the effect of swelling and thermal ageing
CN245-5 PPT-5	M. Veshchunov	IAEA	IAEA activities in the area of Nuclear Power Reactor Fuel Engineering
CN245-94 PPT-94	A. Kozlov	Russian Federation	Examination of Fast Reactor Materials and Structural Elements at JSC “INM” Premises

5.5.4. Session 5.4. Advanced Fast Reactor Fuel Development – II

Session 5.4. comprised three presentations from the Russian Federation related to the fuel pins of BN-600 reactor.

A. Kozlov (Russian Federation) presented the main results on modelling the different stages of structural changes in austenitic steels, radiation-induced swelling in particular, as well as the effect of structural changes on physical and mechanical properties of steels produced under irradiation in fast reactors. The presentation was about ongoing activities on the development of models of changes in austenitic steel structure and properties under irradiation in fast reactors. Their applicability to the BN-600 reactor claddings was demonstrated.

E. Kinev (Russian Federation) presented that the electrical potential testing proved itself as a rapid-method for the cladding state evaluation after operation in BN-600 reactor. The presentation showed a theoretical model and experimental results that demonstrate a correlation between material radiation-induced swelling, cladding corrosion thinning, and the change of electrical resistivity.

S. Belov (Russian Federation) presented the results obtained in the tests performed with EK164 steel as a cladding material for a high burnup of oxide fuel (UO_2, MOX). The presentation described the activities aimed at improving the quality of cladding tubes both in the stage of fuel rod cladding manufacture and in the metallurgic stage of tubing stock manufacture. The EK164 steel is a proved advanced material with increased radiation resistance, which is confirmed by irradiation examination experience. Planned irradiation examinations of FAs with EK164 claddings would allow obtaining experimental data to justify elongation of fuel life and increase of fuel burnup in BN-600, BN-800 reactors, and application of obtained data to justify operability of BN-1200 fuel elements of the initial operation stage.

TABLE 4. PRESENTATIONS FROM SESSION 5.4. – ADVANCED FAST REACTOR CLADDING DEVELOPMENT - II

Chair: **V. Chuyev** and **T. Asayama**

Id	Presenter	Country	Title
CN245-95 PPT-95	A. Kozlov	Russian Federation	Modelling of Processes in Austenitic Steel Produced Under Irradiation in Fast Reactors and Possibilities of Model Practical Application
CN245-107 PPT-107	V. Shikhalev E. Kinev	Russian Federation	Preliminary Inspection of Spent Fast Reactor Fuel Claddings
CN245-409 PPT-409	S. Belov	Russian Federation	Operability validation of fuel pins with claddings made of EK164-id steel in the BN-600 reactor

5.5.5. Session 5.5. Large Component Technology – I

Five papers were presented in Session 5.5.

K. Vulliez (France) presented a paper on Experimental Qualification of Rotatable Plug Seals for Sodium Fast Reactor on a Large-Scale Test Stand. The presentation detailed a very innovative solution compared with the previous option (i.e. liquid metal seal (tin–bismuth, etc.). A qualification procedure was discussed, including a detailed description of the facility. The target was to obtain a leakage rate below 0.01 Nl/h. The loss of mass for a movement corresponding to 100 Km is negligible, without production of a significant quantity of particles. No deleterious effects due to sodium aerosols are anticipated in the event of contact. Irradiation is not foreseen as having any significant effect on mechanical properties. The future goals of the programme of qualification were clearly described including the characterization of potential ageing effects.

H. Kim (Republic of Korea) delivered a presentation on Heat Transfer Performance Test for a Sodium to Air Heat Exchanger with an Inclined Finned-Tube Bank and described an innovative design for a sodium air heat exchanger for the decay heat removal system. Respective experimental testing on SELFA loop and modelling were also described in order to consolidate the heat transfer performance. There was a reasonably good agreement between experimental and calculated results, even though these calculated values were sometimes slightly underestimated.

S. Rukhlin (Russian Federation) delivered a presentation entitled Development of the Built-in Primary Sodium Purification System, described the development of the built-in primary sodium purification system for BN-1200 and detailed a comparison between integrated and conventional external options. One of the most important advantages claimed for the ‘integrated’ option is the avoidance of active sodium circulation out of the primary vessel. However, the complexity of the design is underscored, mainly due to the need to anticipate various operating conditions during nominal operation or shutdown conditions, including variable impurity levels. The coolants suggested for the cold trap are argon, sodium and gallium. However, sodium poses its own difficulties to implementation owing to its high melting temperature (97.8°C). The anticipated life duration prior to removal of impurities is more than ten years. It was also stressed that the operational feedback from the Superphenix integrated primary purification system was very positive with regard to various operating conditions, including a serious pollution event which occurred in June 1990.

P. Pillai (India) delivered a presentation on the design of a sleeve valve mechanism for the primary sodium pump of a future FBR in India with a 600 MW(e) capacity. The sleeve can be raised or lowered using three tie-rods, and a universal coupling is provided in the tie-rods.

D. Plancq (France) delivered a presentation on ASTRID French SFR: Progress in Sodium Gas Heat Exchanger Development. The details were given on the development of the sodium-gas (nitrogen) heat exchanger dedicated to the Brayton cycle based alternative energy conversion system for ASTRID, including the qualification strategy on DIADEMO facility (40 kW). A potential issue was suggested, namely, the influence of nitrogen on mechanical properties, more particularly at the interface with the cover gas, referring to studies carried out during the 1960s. It was assumed that this potentially deleterious effect only arises at temperatures higher than 500°C, the current operating temperature of this component.

TABLE 5. PRESENTATIONS FROM SESSION 5.5. – LARGO COMPONENT TECHNOLOGY – I

Chair: **B. Margolin** and **C. Latge**

Id	Presenter	Country	Title
CN245-24 PPT-24	K. Vulliez	France	Experimental qualification of rotatable plug seals for Sodium Fast Reactor on a large-scale test stand
CN245-183 PPT-183	H. Kim	Korea, Republic of	Heat Transfer Performance Test for a Sodium-to-Air Heat Exchanger with an Inclined Finned-Tube Banks
CN245-404 PPT-404	S. Rukhlin	Russian Federation	Development of the built-in primary sodium purification system for the
CN245-325 PPT-325	P. Pillai	India	Design of Sleeve Valve mechanism for Primary Sodium Pump of future FBR
CN245-286 PPT-286	D. Plancq	France	ASTRID French SFR: Progress in Sodium Gas Heat Exchanger development

5.5.6. Session 5.6. Liquid Metal Technologies

Session 5.6. comprised four presentations, three from France and one from the Russian Federation.

M. Blat-Yrieix (France) presented an overview of stainless steels susceptibility to different corrosion mechanisms during maintenance operations: Stress corrosion cracking (SCC) induced by caustic solution and inter granular attack (IGA) induced by acid solution used during maintenance operation; the feedback and lessons learned from Phenix operation and maintenance operation, as well as the current opportunity to investigate and characterize materials coming from the Phenix dismantlement. The precautions for ASTRID design and future operation were highlighted.

T. Cozzika (France) presented part of the R&D carried out on components from Phenix fast breeder reactor during its ongoing dismantling operations. Thorough modelling and the investigations of these components will help to predict the compatibility with sodium for the components of the future Gen IV ASTRID prototype reactor. Substantial amounts of valuable information regarding Phenix NPP component materials such as austenitic stainless steels can be gained, since the base metal or welds subjected to normal and abnormal service conditions are difficult to reproduce in the laboratory. In 2012, the first sampling was performed on the Phenix CPML0373 rod made of 304L and 316L austenitic stainless steel which had been exposed to sodium at high temperature for about twelve years effective full power. Examinations (SEM, EDX, XRD) on the Phenix rod are underway and there will be an opportunity to improve overall understanding of sodium coolant chemistry and its interactions with materials.

M. Girard (France) presented results from the testing of an electrochemical hydrogen meter (ECHM) developed in IGCAR, India, in a sodium facility at CEA in France. ECHM which works in equilibrium mode provides an alternate technology to the conventional diffusion-based hydrogen sensor (SPHYNX) that works in dynamic mode for detecting steam leaks into sodium. Tests involved introducing sodium hydride in liquid sodium to change the hydrogen concentration in sodium. The signals of both ECHM and SPHYNX were monitored as a function of hydrogen concentration and temperature. The test results showed good agreement between the output of both types of sensors in detecting the change in hydrogen concentration in sodium and the response time of ECHM was found to be higher than SPHYNX.

E. Varseev (Russian Federation) presented numerical simulation of a sodium cold traps performance characteristics. The method of consecutive heat and mass transfer simulation using custom open source CFD solver to determine the mass transfer and retain coefficients for the cold trap mock-up was discussed. The new solver has been developed, which calculates fields of dissolved oxygen and suspended sodium oxide particles and the rate of their accumulation in the cold trap as well. It had been demonstrated that suggested methodology of CFD simulation allows the analysis of operational parameters of sodium purification systems for the nuclear facilities.

The presentations were followed by the discussion where participants discussed i) measures of hydrogen in sodium by means of the electrochemical sensors and ii) specific features of mass transfer simulation model. In addition, the participants exchanged experiences in the experimental studies of stainless steels corrosion in sodium.

TABLE 6. PRESENTATIONS FROM SESSION 5.6. – LIQUID METAL TECHNOLOGIES

Chair: **R. Askhadullin** and **A. Yamaguchi**

Id	Presenter	Country	Title
CN245-238 PPT-238	M. Blat-Yrieix	France	Stainless Steels Corrosion in Sodium Fast Reactor: Feedback from Risks during Maintenance Operations (SCC in Caustic Solution and Intergranular Corrosion by Acid Solution)
CN245-349 PPT-349	T. Cozzika	France	Chemical compatibility with liquid sodium after in service solicitations: feedback on stainless steel in French sodium Fast reactor after 35 years of operation
CN245-489 PPT-489	M. Girard	France	Testing of electrochemical hydrogen meter in a sodium facility in Cadarache
CN245-163 PPT-163	E. Varseev	Russian Federation	Mass Transfer Simulation Model for Justification Sodium Purification System Characteristics

5.5.7. Session 5.7. Chemistry Related Technology

The session comprised six papers on chemistry related technologies for liquid metals, water and gas.

A. Legkikh (Russian Federation) introduced the methods for controlling the oxygen concentration in heavy liquid metal (lead and lead–bismuth) in nuclear reactors and test facilities. He presented several technologies and noted the following conclusions:

- The inert gas–oxygen gas mixture method is efficient and convenient in small test facilities (up to 10 L) with static liquid metal;
- The hydrogen–water (steam)–inert gas mixture method can be effectively used for coolant oxidation in research facilities having isothermal operating conditions and experimental installations without forced circulation of coolant; and
- The solid phase method has been indicated as the most advanced method to control the oxygen concentration in non-isothermal loops with forced convection.

C. Fazio (EC) gave an overview on corrosion of the materials in the fast reactor environment, focussing on the LFR and SFR. Ms Fazio highlighted the importance of chemistry control and, in particular, the oxygen control to assess the corrosion of structural materials when exposed to the liquid metals. There were the following conclusions:

- Materials corrosion needs attention since it can cause potential loss of load bearing capability and loss of mechanical strength of the components;
- Corrosion in liquid sodium seems to be less severe with respect to Heavy Liquid Metal (HLM);
- Corrosion mitigation options that have been investigated are strictly related to the chemistry of the liquid metal; and there are known technologies with regards to sodium, although further improvement should not be excluded as they are still under investigation with regards to HLM. However, knowledge of the 'chemistry' of the liquid metals is essential for corrosion control;
- A step forward is needed for both sodium and HLM for evolving from experimental and empirical assessment to physical models.

A. Aerts (Belgium) presented the conditioning and chemistry programme for MYRRHA. MYRRHA is an LBE cooled 100 MW(th) accelerator driven system under development at SCK•CEN. The MYRRHA chemistry control programme has essentially three components: oxygen control and monitoring, coolant impurity management and radionuclide release and capture. The author summarized the achievement of the chemistry control programme as follows:

- Accurate and precise oxygen sensors for LBE cooled systems were developed and tested;
- Oxygen control with a solid phase method was accomplished on both laboratory and pilot scales. Moreover, electrochemical oxygen pumps were also developed and tested;
- By experiment and theory, a scientific basis has been established to assess the behaviour of impurities (corrosion products, etc.) in LBE, which guides the development of coolant purification systems;

- A global thermochemical model has been developed that describes the complex radionuclide chemistry in the LBE and progress has been made in understanding radionuclide/polonium chemistry and predicting release from the LBE.

The contribution of **R. Askhadullin (Russian Federation)** on strategies for maintaining appropriate technology of heavy liquid coolants in advanced nuclear power plants was presented by A. Legkikh who discussed the following items:

- The design and building of coolant technology methods and tools, and their direct implementation are needed for coolant preparation, power plant start-up, lifetime operation and decommissioning;
- The coolant technology methods and tools, which are under development for the power plant, include the following components: (i) hydrogen purification of the coolant and circuit from slag forming impurities, (ii) dissolved oxygen control in the coolant to ensure corrosion protection of steels, (iii) coolant and cover gas filtration unit, (iv) coolant purification from impurities, and (v) a system of coolant control in both reactor and non-reactor conditions;
- A key point that ensures the reliable operation of future HLM reactor systems is the training of specialists in HLM coolant technology.

V. Yurmanov (Russian Federation) (representing **K. Shutko**) presented the development of steam–water cycle chemistry for the steam generator of the MBIR. V. Yurmanov indicated in the presentation that neutral water chemistry had been proposed for the SG of the MBIR for the following reasons:

- Simple chemistry control and monitoring and reduced capital cost due to the absence of any chemical reagent dosing into feedwater;
- The absence of hydrazine and ammonia dosing eliminates both toxicological hazards for personnel and the required exchange capacity of ion exchangers; and
- Elimination of deposits from SG surfaces during operational transients.

The contribution concluded with a suggestion to the IAEA to develop a water chemistry control programme for LMC fast reactors similar to the SSG-13 (2011\2014) issued for LWRs and to improve the current edition of LMC guidelines (NP-T-1/6 (2012)).

L. Bělovský (Czech Republic) contributed to the session with a presentation on helium recovery from the guard vessel atmosphere of the ALLEGRO reactor. The ALLEGRO reactor is a helium cooled fast reactor under development within Europe to demonstrate the viability of the GFR technology. ALLEGRO has a specific design feature which is the close containment–guard vessel. A potential deployment of GFRs would require large quantities of helium, therefore, a key requirement of the GFR is to address the helium economy (recovery of expected helium leakage into the guard vessel). The question addressed in this contribution is related to the technical and economic viability of the recovery of leaked helium from a N_2+He mixture. The studies performed had shown that the membrane technology seems to be suitable to answer this question. However, membrane separation can be further optimized. The next step would be to perform a small-scale experiment to demonstrate the scalability of the complex helium recovery system.

TABLE 7. PRESENTATIONS FROM SESSION 5.7. – CHEMISTRY RELATED TECHNOLOGY

Chair: **A. Legkikh** and **C. Fazio**

Id	Presenter	Country	Title
CN245-392 PPT-392	A. Legkikh	Russian Federation	Methods of controlling concentration of oxygen dissolved in heavy liquid metal coolants (lead and lead-bismuth) of nuclear reactors and test facilities
CN245-562 PPT-562	C. Fazio (Invited)	EC	Materials corrosion in Fast Reactor environment
CN245-211 PPT-211	A. Aerts	Belgium	The Conditioning and Chemistry Programme for MYRRHA
CN245-393 PPT-393	R. Askhadullin	Russian Federation	Strategies of maintaining appropriate technology of heavy liquid metal coolants in advanced nuclear power plants
CN245-543 PPT-543	V. Yurmanov	Russian Federation	Development of steam-water cycle chemistry for steam generator of research reactor MBIR
CN245-390 PPT-390	L. Bělovský	Czech Republic	Helium Recovery from Guard Vessel Atmosphere of the ALLEGRO Reactor

5.5.8. Session 5.8. Structural Materials

This session consisted of five presentations.

S. Kim (Republic of Korea) delivered a presentation on fabrication and evaluation of the advanced cladding tube for the PGSFR. The author noted that KAERI has developed new cladding materials and fabricated the cladding tubes with new cladding materials (FC92). Out of pile tests had been carried out and the irradiation test of the cladding tube had been performed in BOR-60.

B. Margolin (Russian Federation) discussed the basic principles for lifetime and structural integrity assessment for the BN-600 and BN-800 fast reactors' components. The main mechanism of embrittlement (due to hardening, swelling and phase transformation) for materials used for BN fast reactor components were reviewed. An algorithm is proposed for estimation of the structural integrity assessment, taking into account the above mechanisms.

A. Courcelle (France) delivered a presentation on the Transmission Electron Microscopy (TEM) characterization of a swell resistant austenitic steel irradiated at high temperature in the PHENIX fast reactor. The results of TEM examination for austenitic 15Cr-15Ni-Ti-Mo steel after high temperature irradiation indicated that there is precipitation of G-phase, Laves-phase and helium bubbles on grain boundaries.

C. Sowrinathan (India) reported on the performance evaluation of ferroboron shielding material after irradiation in the FBTR reactor. Ferroboron, which is a candidate material for in-vessel radiation shielding application, was irradiated in FBTR. After the irradiation test, neutron radiography, released helium measurement and metallography were carried out. Test results had shown that ferroboron is a suitable material for deployment in future fast reactors.

A. Buchatsky (Russian Federation) delivered a presentation on the prediction of creep–rupture properties for austenitic stainless steel that has undergone neutron irradiation at different temperatures. To verify the physical and mechanical model for austenitic stainless steel, in-reactor tests with a gas filled tube in the RBT-6 were carried out. There was a good agreement between the results calculated by the model and the experimental data.

L. Nicolas (France) discussed the recent supply of 316L(N) stainless steel product for ASTRID reactor. The test results (mechanical tests and microstructural examination) for material of blanks of different structural components for ASTRID, including seamless tubes, were presented. It was shown that most of the results satisfy the requirements.

TABLE 8. PRESENTATIONS FROM SESSION 5.8. – STRUCTURAL MATERIALS

Chair: **A. Sorokin and S. Kim**			
Id	Presenter	Country	Title
CN245-33 PPT-33	S. Kim	Korea, Republic of	Fabrication and Evaluation of Advanced Cladding Tube for PGSFR
CN245-130 PPT-130	B. Margolin	Russian Federation	Basic principles for lifetime and structural integrity assessment of BN-600 and BN-800 fast reactors components with regard for material degradation
CN245-135 PPT-135	A. Courcelle	France	TEM characterization of a swelling-resistant austenitic steel irradiated at high temperature (>600°c) in the PHENIX fast reactor
CN245-250 PPT-250	C. Sowrinathan	India	Performance evaluation of ferroboron shielding material after irradiation in FBTR
CN245-92 PPT-92	A. Buchatsky	Russian Federation	Prediction of creep-rupture properties for austenitic stainless steels undergone neutron irradiation at different temperatures
CN245-477 PPT-477	L. Nicolas	France	Recent supplying of 316L(N) stainless steel products for ASTRID

5.5.9. Session 5.9. Large Component Technology – II

Session 5.9. comprised two presentations, one from China and one from India.

F. Gao (China) presented the development of a computer code for predicting fast reactor oxide fuel element thermal and mechanical behaviour (FIBER-oxide). The purpose, physical models and test results of the computational code were explained.

V. Rajan Babu (India) discussed the practical issues arising during the manufacture of the reactor components. There were many fabrication challenges, which were well-defined in this presentation, especially those pertinent to welding. All manufacturing was performed domestically.

TABLE 9. PRESENTATIONS FROM SESSION 5.9. – LARGE COMPONENT TECHNOLOGY - II

Chair: **J. Kuzina** and **H. Kim**			
Id	Presenter	Country	Title
CN245-217 PPT-217	F. Gao	China	The development of a computer code for predicting fast reactor oxide fuel element thermal and mechanical behaviour (FIBER-Oxide)
CN245-510 PPT-510	V. Rajan Babu	India	Challenges During Manufacture of Reactor Components of PFBR

5.5.10. Session: 5.10. Fuel Modelling and Simulation

This session consisted of five presentations, three from the Russian Federation and two from France.

E. Marinenko (Russian Federation) from Institute for Physics and Power Engineering, JSC "SSC RF – IPPE", discussed on the development of the DRAKON code designed for numerical simulations of temperature and stress-strain state of fast reactors nitride fuel pins, and its verification based on post irradiation evaluation (PIE) data obtained after irradiation of standard fuel assemblies (FA) in BR-10 reactor (with uranium mononitride), in BOR-60 reactor (experiment BORA-BORA) and in BN-600 reactor (with mixed nitride fuel). Within the framework of the PRORYV project, a comprehensive programme for calculation and experimental studies of mixed nitride fuel for BN-1200 and BREST-OD-300 reactors has been designed.

B. Michel (France) from CEA, Fuel Studies Department, presented an approach based on 3D simulations of fuel fragment behaviour. The goal of the study was to investigate mechanisms contributing to the fuel-to-cladding gap closure at beginning of life (BOA) reactor state. The computations involved a chaining of two simulation tools developed in PLEIADES software environment: a first calculation was made by fuel performance code GERMINAL, to fix the evolutions of physical variables in fuel during the irradiation scenario to be considered. Using these evolutions as loadings, the second 3D calculation was performed with LICOS software. The basic assumptions of the 3D modelling were discussed. The interpretation of the calculation results showed that thermal-induced hourglass shape of the fragment and mass transfer are both contributing to the fuel-to-cladding gap closure. As working perspectives, a coupled formulation of mechanics and fuel restructuring would be expected.

M. Lainet (France), also from CEA, Fuel Studies Department, presented the current status and progression of GERMINAL fuel performance code for SFR oxide fuel pins. GERMINAL is developed within PLEIADES software environment. It has been validated and is now currently used for ASTRID design studies. Modelling evolutions are continuously integrated in the code. As a first example, a revision of the fuel fragments relocation model, linked to author's previous study was presented. For a better rendering of thermal-induced effects, the average temperature gradient inside pellet has been adopted as new driving parameter of the model. A second modelling evolution concerned a coupling with OpenCalphad thermochemistry component. By involving thermodynamic equilibrium calculations in fuel and gap, the goal was to improve the evaluation of the amount of fission products phases that contribute to the Joint Oxyde-Gaine formation, after release. Further works for GERMINAL would be related to the implementation of a coupling with a fission gas behaviour component, the extension of the validation base and the development of next major version, that would allow parallel calculations.

A. Boldyrev (Russian Federation) from IBRAE, presented the status of the fast reactor fuel performance code BERKUT developed within the PRORYV project for fuel rods with mixed oxide and nitride fuels. An advanced mechanistic approach applied in the code for modelling of fuel microstructure evolution, swelling, fission product release and thermo-chemical transformations allowed satisfactory description of thermo-mechanical and physic-chemical behaviour of fuel rods under irradiation in reactors BOR-60 and BN-600 and was in line with the results of post irradiation examinations (PIE) of fuel rod samples.

I. Golovchenko (Russian Federation) from RIAR, discussed a wide experience and applicability of high density metal uranium in advanced BN-reactors. Potential advantages of the novel fuel for fast reactors were demonstrated, comparison with more traditional fuels was presented and thoroughly discussed.

TABLE 10. PRESENTATIONS FROM SESSION 5.10. – FUEL MODELLING AND SIMULATION

Chair: **M. Veshchunov** and **M. Lainet**

Id	Presenter	Country	Title
CN245-63 PPT-63	E. Marinenko	Russian Federation	Problems of calculation modelling of nitride fuel performance: DRAKON code
CN245-223 PPT-223	B. Michel	France	3D simulation in the PLEIADES software environment for sodium fast reactor fuel pin behaviour under irradiation
CN245-222 PPT-222	M. Lainet	France	Current status and progression of GERMINAL fuel performance code for SFR oxide fuel pins
CN245-363 PPT-363	A. Boldyrev	Russian Federation	BERKUT – Best Estimate Code for modelling of Fast Reactor Fuel Rod Behaviour under Normal and Accidental Conditions
CN245-396 PPT-396	I. Golovchenko	Russian Federation	Experience and applicability of high density metal uranium in advanced BN-reactors

5.5.11. Track 5. Poster Session

Track 5 comprised a total of 55 poster presentations.

Considerable new information on the topics related to the track was presented. The most important point concerned actual information exchange between representatives of different national programmes. In many cases, these programmes have important essentials and open discussion during the sessions helped everybody to gain a better understanding.

For example, there is not the same level of understanding with respect to preferred fuel type for FBRs: metallic, MOX, mixed carbides and mixed nitrides. Every participant had their own reasons, which were fruitfully discussed.

A similar situation concerns to the use of liquid metal coolant (sodium, lead and lead–bismuth) technologies, which were discussed along with reasons behind the selection of the particular coolant.

5.6. TRACK 6 – FAST REACTORS, EXPERIMENTS, MODELLING AND SIMULATIONS

5.6.1. Session 6.1. CFD and 3D Modelling

Six papers were presented in this session, with three papers dealing with three-dimensional computational fluid dynamic (CFD) code developments as well as applications of the commercial CFD codes for fast reactor thermohydraulics.

D. Fomichev (Russian Federation) demonstrated the capabilities of the commercial CFD code STAR-CCM+ and the in-house developed CFD code LOGOS to predict thermohydraulic characteristics of lead coolant flow in BREST-300 fuel subassembly. Pressure drop values on the spacer grid estimated using empirical correlations were found to differ by ~20% from those predicted by the CFD codes. However, a good agreement was found between friction factors of the rod bundle and the data obtained from the available empirical correlations. Analysis of the heat transfer characteristics of the lead coolant flow indicated that eddy viscosity based two-equation turbulence models are capable of predicting the heat transfer coefficient (Nusselt number) within ~3%.

V. Chudanov (Russian Federation) presented a paper that dealt with the direct numerical simulation (DNS) CFD code CONV-3D, running in cluster computing mode and its application for simulation of (i) thermal stratification of sodium in the upper plenum of the MONJU and BN-600 reactors, (ii) sodium experiments conducted on a model of the Phenix pipe bend and (iii) LBE flow/heat transfer in a 19-pin bundle with wire spacers. Good agreement between the numerical predictions and the experiments was observed that allowed to suggest and that CONV-3D with high predictive power can be used for reactor applications and verification of CFD RANS codes.

H. Ohshima (Japan) elaborated the mathematical modelling aspects of a FEM-based SPIRAL code for the rod bundle thermohydraulics. This included the newly developed hybrid turbulence model coupled with the streamline upwind Petrov-Galerkin method and balancing tensor diffusivity method. Verification of the SPIRAL code against the DNS code was performed. The code has been ‘parallelized’ using MPI for enhancing simulation efficiency.

V. Karuppanna (India) described the development of a steady state one-dimensional model for the once-through type steam generators of fast reactors. Predicted results of the one-dimensional model were compared against the in-house test data of a 5 MW steam generator test facility and satisfactory agreement was observed in the global parameters.

V. Arasappan (India) dealt with solid modelling of the fuel handling system of an Indian PFBR and development of operator training simulator.

All the papers generated fruitful discussions.

TABLE 1. PRESENTATIONS FROM SESSION 6.1. – CFD AND 3D MODELLING

Chair: **V. Karuppanna** and **L. Bolshov**

Id	Presenter	Country	Title
CN245-548 PPT-548	D. Fomichev (Invited)	Russian Federation	Numerical simulation of hydraulics and heat transfer in the BREST-OD-300 LFR fuel assembly
CN245-332 PPT-332	V. Karuppanna	India	Steady State modelling and Validation of Once Through Steam Generator
CN245-274 PPT-274	V. Chudanov	Russian Federation	Applications of the DNS CONV-3D Code for Simulations of Liquid Metal Flows
CN245-378 PPT-378	V. Arasappan	India	3D modelling of Fuel Handling System for PFBR Operator Training Simulator
CN245-453 PPT-453	H. Ohshima	Japan	Numerical Simulation Method of Thermal Hydraulics in Wire-wrapped Fuel Pin Bundle of Sodium cooled Fast Reactor
CN245-99 PPT-99	Z. Tian	China	Modelling and Simulation of Heat Transport System and Steam Power Transition System of CEFR

5.6.2. Session 6.2. Thermal Hydraulics Calculations and Experiments

In Session 6.2., six presentations were given by representatives of Argentina, France, India, Italy, the Republic of Korea and the Russian Federation. Most of the papers were devoted to the calculation results obtained using CFD, sub-channel, coupled one-dimensional, 3D and thermomechanical codes.

J. Hong (Republic of Korea) from KAERI included the numerical simulation of the steam generator of the Prototype Gen IV Sodium cooled Fast Reactor (PGSFR). The multi-dimensional sodium temperature distribution at the tube bundle region was calculated using the STAR-CCM + CFD package, the heat flux from the sodium-side to the water-side was estimated using the one-dimensional in-house code (HSGSA) and supplied as boundary conditions at tube walls in the multi-dimensional CFD simulation. The thermohydraulic analysis results would be provided as the input data to evaluate the mechanical and structural integrity of the steam generator of the PGSFR.

C. R. C. Sowrinathan (India) from IGCAR provided an insight into the behaviour of Mark I carbide fuel for the current operating conditions of the FBTR and the influence of inlet temperature and operating linear heat rating on the achievable burnup.

F. Lodi (Italy) from University of Bologna, presented the results of the sub-channel analysis code (ANTEO+) validation against experiments performed with heavy liquid metal coolants. Several new experimental lines of enquiry have been considered. The results of the validation confirmed the predictive ability of the code, except for the sub-channels and the pins in close proximity to the wrapper.

A. Sorokin (Russian Federation) from JSC IPPE, provided the results of experimental measurements of local coolant velocities in the upper chamber (plenum) and the temperatures in the reactor pool which were taken in the integral water experimental facility simulating high powered fast reactor (scale ~1:10) behaviour in the steady-state forced circulation regime and simulating a nominal operation regime. It had been theoretically proven, that the experimental data obtained on water can be reliably extended to the liquid metal behaviour. The data gained underscore the necessity of taking into consideration the stratification phenomena.

M. L. Japas (Argentina) from CNEA, discussed on the corrections to the sodium density relations formulated by Fink and Leibowitz at high temperatures.

Y. Gorsse (France) from CEA, covered the coupling between CATHARE (STH), TrioMC (sub-channel) and TrioCFD codes that had been developed at CEA. Those codes were integrated into a new code: MATHYS (Multiscale ASTRID Thermal-HYdraulics Simulation). Within MATHYS, TrioMC and TrioCFD are coupled at the boundaries using a domain decomposition approach. Then, the two codes are coupled with a CATHARE using a domain overlapping method. The resulting multiscale simulation tool is able to account for feedback effects between all three scales. The validation results of MATHYS were discussed on existing experiments: TALL-3D for STH/CFD, PLANDTL-DHX for sub-channel/CFD and PHENIX at the integral scale.

The main conclusions of the session discussions were:

- Numerical simulations are extensively used to define or confirm the design and operating characteristics of the LMR to enhance their safety and economic attractiveness;
- CFD codes (mainly proprietary) are widely used for investigation of thermohydraulic processes taking place in the different parts of the reactor facility, though they are not extensively validated against experiments performed with liquid metal coolants. Meanwhile, it is shown that standard turbulence models used in CFD codes do not allow description of the heat transfer processes with high accuracy;
- To describe the multi-dimensional and two-phase thermohydraulic phenomena in the different regimes of LMR operation and to perform extensive calculations in a reasonable time difference coupling methodology of 1-D and 3-D thermohydraulic codes have been developed. The validation results have shown that such an approach is quite reasonable; and
- Coolant properties data evaluation is an important task, in particular, to obtain reliable data for the thermohydraulic calculations.

TABLE 2. PRESENTATIONS FROM SESSION 6.2. – THERMAL HYDRAULICS CALCULATIONS AND EXPERIMENTS

Chair: **S. Perumal** and **N. Mosunova**

Id	Presenter	Country	Title
CN245-249 PPT-249	J. Hong (Invited)	Korea, Republic of	Thermal Hydraulic Study of Steam Generator of PGSFR
CN245-310 PPT-310	C. Sowrinathan	India	Effect of inlet temperature and operating linear heat rating (LHR) on the maximum achievable burnup of MK-1 carbide fuel in FBTR
CN245-232 PPT-232	F. Lodi	Italy	Extension to Heavy Liquid Metal coolants of the validation database of the ANTEO+ sub-channel code
CN245-440 PPT-440	M. L. Japas	Argentina	Density of sodium along the Liquid-Vapour Coexistence Curve, including the Critical Point
CN245-439 PPT-439	A. Sorokin	Russian Federation	Experimental investigations of velocity and temperature fields, stratification phenomena in an integral water model of fast reactor in the steady state forced circulation
CN245-455 PPT-455	Y. Gorsse	France	Development and Validation of Multi-scale Thermal-Hydraulics Calculation Schemes for SFR Applications at CEA

5.6.3. Session 6.3. Neutronics I

Session 6.3. comprised of five presentations, two from the Russian Federation, and one each from France, Mexico and OECD/NEA.

G. Rimpault (CEA, France), presented the APOLLO3 scientific tool for SFR neutronic characterization: current achievements and perspectives. ASTRID is an SFR design that will be France's flagship Gen IV reactor. Its innovative core contains many axial and radial heterogeneities (in order to obtain a negative sodium void reactivity coefficient) and interfaces that are challenging for current deterministic codes to simulate correctly. Hence there is the need for new improvements in modelling (3D simulations, parallel processing) like those being elaborated within the APOLLO3 platform. The APOLLO3-SFR package built with APOLLO3 solvers defines reference calculation schemes associated with a nuclear data library to calculate all neutronic parameters (critical masses, sodium void, Doppler coefficient, β_{eff}, etc.) together with certified biases and uncertainties derived from the VV&UQ process. This VV&UQ process incorporates numerical validation, a-priori uncertainties based on nuclear data covariances as well as experimental validation mainly from MASURCA, a fast mock-up reactor, located at CEA Cadarache. A future programme called GENESIS will be performed in support to the prototype ASTRID to validate the CFV core (low void core) specificities such as sodium void reactivity, control rod worth, power map distribution. A part of the GENESIS experimental programme contains integral experiment underway at the BFS facility.

O. Andrianova (Russian Federation), presented integral experiments with minor actinides (MA) at the BFS critical facilities: state of the art survey, re-evaluation and application. Improvements of MA nuclear data to increase the accuracy of reactor characteristics require, on a par with the further supplementation of experimental databases with new neutron cross-section differential measurements, a wider use and accumulation of information about the integral measurements. The feasibility of creating neutron spectra similar to those of the reactors under study increases the practical value of integral experiments since it makes it possible to reflect the MA properties, which are characteristics of a given spectrum and thus reduces the calculation uncertainty of reactor performance caused by the MA nuclear data uncertainty. The formed database on MA measurements at the BFS facilities can be used to test and adjust evaluated nuclear data files, and verify calculation codes. The results of the analysis of all the MA integral experiments can be a starting point for planning further BFS experimental programs on the study of MA characteristics in fast reactors.

V. Pershukov (Russian Federation) presented on International Research Centre based on MBIR reactor – cornerstone for Gen IV technologies development. It was discussed that the MBIR reactor would have the unique technical and experimental capabilities that should facilitate the future studies in justification of advanced reactor technologies to be ensured after the year 2020. International collaboration was formed to ensure effective utilization of the MBIR unique features and to choose and install the needed experimental, scientific and processing equipment, as well as measuring instruments. International cooperation within the International Research Centre based on MBIR reactor, would contribute to the development of the technologies of nuclear fuel cycle closure in 21st century nuclear power engineering using fast neutron reactors. Formulation of the multilateral scientific programme by the advisory board would allow to share the risks and costs between the participants.

A. Gomez Torres (Mexico) presented on verification of the neutron diffusion code AZNHEX by means of the Serpent-DYN3D and Serpent-PARCS solution of the OECD/NEA SFR Benchmark. AZNHEX is a neutron diffusion code for hexagonal-z geometry currently under development as part of the AZTLAN project in which a Mexican platform for nuclear core simulations is being developed. The main objective of the presented work was to test the AZNHEX code capabilities against two well-known diffusion codes DYN3D and PARCS. The AZNHEX results showed some higher discrepancy from Serpent than DYN3D and PARCS, however, such discrepancy can be considered acceptable for a novel development. It can be concluded that the AZNHEX obtained results compared very well against the other well validated and known codes DYN3D and PARCS and also while comparing against the stochastic code Serpent. The resulting differences were small enough to consider the comparison a success, but there would be still work to do in order to further verify and validate the AZNHEX code.

T. Ivanova (OECD/NEA), presented on benchmark evaluation of Dounreay prototype fast reactor minor actinide depletion measurements. An international collaboration between Japan, the USA, and the UK has been established through the OECD/NEA to evaluate the PFR minor actinide depletion measurements as benchmarks for inclusion in the international reactor physics experiment evaluation project (IRPhEP) handbook and Spent Fuel Isotopic Composition (SFCOMPO) database. These experiments offer a unique set of measurements for the evaluation of nuclear data needed to support modern neutronic simulation of fast reactors and their accompanying fuel cycle activities to collate detailed information, including uncertainties, regarding PFR and on the experiments underway.

TABLE 3. PRESENTATIONS FROM SESSION 6.3. – NEUTRONICS I

Chair: **A. Rineiski** and **D. Klinov**			
Id	Presenter	Country	Title
CN245-216 PPT-216	G. Rimpault (Invited)	France	The APOLLO3 scientific tool for SFR neutronic characterization: current achievements and perspectives
CN245-10 PPT-10	O. Andrianova	Russian Federation	Integral Experiments with Minor Actinides At The Bfs Critical Facilities: State Of The Art Survey, Reevaluation And Application
CN245-578 PPT-578	V. Pershukov (Invited)	Russian Federation	International research centre based on MBIR reactor – cornerstone for Gen IV technologies development
CN245-397 PPT-397	A. Gomez Torres	Mexico	Verification of the neutron diffusion code AZNHEX by means of the Serpent-DYN3D and Serpent-PARCS solution of the OECD/NEA SFR Benchmark
CN245-142 PPT-142	T. Ivanova	OECD/NEA	Benchmark Evaluation of Dounreay Prototype Fast Reactor Minor Actinide Depletion Measurements

5.6.4. Session 6.4. Neutronics – II

This session was comprised of four presentations.

B. Vrban (Slovakia) from B&J Nuclear presented a stability analysis of the Republic of Korea prototype reactor PGSFR based on core eigenvalue analysis. A modified version of the DIF3D 10.0 neutron diffusion code was used to compute higher order eigenmodes and was validated against a simple benchmark. The first four PGSFR eigenvalues were calculated as a function of the core diameter to height ratio. The method can be used by core designers to suggest changes towards increased eigenvalue separation, hence more stable configurations. Mr Vrban confirmed that a time synthesis using these static modes in a modal expansion was consistent with the time dependent solution. The author also commented that the small discrepancy between the first and second harmonics was not a consequence of degeneracy (caused by rotational symmetry in the horizontal plane), but due to the fact that the core had actually two nearly identical eigenmodes as a consequence of an almost symmetric control rod pattern.

R. Jacqmin (France) from CEA presented sodium void and control rod worth experiments recently undertaken in BFS-1 as part of a joint CEA–IPPE programme simulating an axially heterogeneous SFR core. The measurements were compared with Monte Carlo calculations performed with different nuclear data sets and codes. Core reactivity, β_{eff}, and spectral indices were well predicted. Discrepancies were noted, however, in some computed-over-experimental values for sodium void reactivity, as well as for control rod worth, especially in voided situations (10% overestimation in magnitude with JEFF-3.2 data). R. Jacqmin explained that the analysis was still ongoing, that the reported results were likely to be revised, and that firm conclusions on the relative performance of the nuclear data libraries could not yet be drawn at this stage.

X. Huo (China) from CIAE presented a subset of experimental data obtained from the CEFR neutronics start-up tests. These data include control rod worth, sodium void reactivity (obtained with a special movable subassembly), temperature effects and foil activation. It was explained that the CIAE would be interested in discussing these CEFR experimental data with international experts as part of an IAEA CRP, in order to determine whether the data could be turned into an international experimental benchmark. The experts would be asked to establish the required level of information to be included in the description of this benchmark.

E. Del Valle Gallegos (Mexico) from NINR presented the results of two OECD SFR neutronics benchmarks (MOX-3600 and MET-1000) calculated with the AZNHEX Hex-Z nodal finite element code. The numerical solution of the diffusion equation uses a hexagonal-to-Cartesian mesh transformation technique. The OECD benchmark calculations were simulated as part of the code V&V. Multigroup cross-sections were generated by separate Monte Carlo simulations (using SERPENT) over two-dimensional or 3D supercells. The computed reactivity was found to be in reasonable agreement with the corresponding reference SERPENT results in the nominal conditions. However, discrepancies were observed in rodded conditions (400 pcm) and in sodium voided conditions (-800 pcm). E. Del Valle commented that the reasons for these discrepancies is yet to be sought.

TABLE 4. PRESENTATIONS FROM SESSION 6.4. – NEUTRONICS – II

Chair: **R. Jacqmin** and **E. Seleznev**

Id	Presenter	Country	Title
CN245-180 PPT-180	B. Vrban	Slovakia	Stability Analysis of a Liquid Metal Cooled Fast Reactor
CN245-470 PPT-470	R. Jacqmin	France	Analysis of the BFS-115-1 experiments
CN245-501 PPT-501	X. Huo (Invited)	China	Physical start-up test of China Experimental Fast Reactor
CN245-411 PPT-411	E. Del Valle Gallegos	Mexico	Solution of the OECD/NEA SFR Benchmark with the Mexican neutron diffusion code AZNHEX

5.6.5. Session 6.5. Uncertainty Analysis and Tools

Session 6.5. comprised of four presentations, two from France, and one each from Italy and the Russian Federation.

A. Gandini (Italy), presented on recent and potential advances of the heuristic generalized perturbation theory (HGPT) methodology. In the presentation two recent advances and a potential one, all based on the heuristic GPT (HGPT) methodology, were presented. The first two advances concerned, respectively, a method for the on-line monitoring of a subcritical (ADS) system and a method for detecting hot spots in a power reactor via prompt response Self Powered Neutron Detectors (SPND). The third one concerned the potential implementation of GPT methods in Monte Carlo codes.

G. Rimpault (France) presented on Evaluation of β_{eff} measurements from BERENICE programme with TRIPOLI4 and uncertainties quantification. The ASTRID project needs a better understanding on the uncertainties of the effective delayed neutron fraction because this parameter is the upper limit of the prompt criticality and sets the safety margins. The use of Monte-Carlo code TRIPOLI4 and its recent development of the iterated fission probability method allow to improve the C/E ratio and give credit to the deterministic code, ERANOS, for calculating β_{eff}. The detailed representation of cores and the use of an energy dependency of the delayed neutron emission to the incident neutron energy were the major contribution to this improvement. Also, the improvement comes from the calculated terms used to derive β_{eff} from raw experimental measurements. The complementary use of the deterministic code ERANOS is fundamental for the uncertainty quantification process. New experimental techniques (noise with faster electronic) as the ones envisaged within the future experimental programme: GENESIS in the recently refurbished facility: MASURCA could reduce current uncertainties of reference codes in calculating β_{eff}.

G. Manturov (Russian Federation), presented a system of codes and nuclear data for neutronics calculations of fast reactors and uncertainty estimation. Designing of neutronics characteristics of fast reactor cores and fuel cycle requires to use certified, qualified sets of codes and nuclear physics constants. As the main calculation uncertainty is connected to the used nuclear physics constants, they should be adequate to the most reliable evaluations of nuclear data. Group constants sets, which are used as input data in nuclear physics engineer calculations, and which are based on those data – they should be validated and certified together with the codes. Nowadays, in many institutions of ROSATOM (Russian Federation), designing of neutronics characteristics of fast reactor cores (as BN, BREST, SVBR, MBIR) is based on ABBN group nuclear data, which have long history beginning from 1962. For treating the ABBN data a special code system CONSYST/ABBN (IPPE) was developed.

G. Rimpault (France) presented on objectives and status of the OECD/NEA sub-group on uncertainty analysis in modelling (UAM) for design, operation and safety analysis of SFRs (SFR-UAM). The sub-group started its work under the NSC/WPRS/EGUAM two years ago and has been meeting every year. The participants to the sub-group launched a series of benchmarks to support current understanding of important phenomena to define and quantify the main core characteristics affecting safety and performance of SFRs. Different codes and data had been used to support the evaluation of the uncertainties which challenges existing calculation methods. Two SFR cores were selected for the SFR-UAM benchmark, a 3600 MW(th) oxide core and a 1000 MW(th) metallic core. Their neutronic feedback coefficients were calculated for transient analyses. The SFR-UAM sub-group had been defining the grace period or the margin to melting available in the different accident scenarios and this within uncertainty margins. Recently, the work of the sub-group has been updated to incorporate new exercises, namely, the depletion benchmark, the control rod withdrawal benchmark, and the Super-Phenix start-up transient. Experimental evidence in support of the studies was also being developed.

TABLE 5. PRESENTATIONS FROM SESSION 6.5. – UNCERTAINTY ANALYSIS AND TOOLS

Chair: **G. Rimpault** and **A. Kiselev**

Id	Presenter	Country	Title
CN245-567 PPT-567	A. Gandini (Invited)	Italy	Recent and Potential Advances of the HGPT methodology
CN245-218 PPT-218	G. Rimpault	France	Evaluation of β_{eff} measurements from BERENICE programme with TRIPOLI4® and uncertainties quantification
CN245-475 PPT-475	G. Manturov	Russian Federation	System of Codes and Nuclear Data for Neutronics Calculations of Fast Reactors and Uncertainty Estimation
CN245-255 PPT-255	J. Heo	Korea, Republic of	Sensitivity and Uncertainty Analysis in Best-Estimate modelling for PGSFR Under ULOF Transient
CN245-220 PPT-220	G. Rimpault	France	Objectives and Status of the OECD/NEA sub-group on Uncertainty Analysis in modelling (UAM) for Design, Operation and Safety Analysis of SFRs (SFR-UAM)

5.6.6. Session 6.6. Coupled Calculations

Session 6.6. comprised of five presentations.

V. Strizhov (Russian Federation) discussed codes of new generation developed for "Breakthrough" (PRORYV) project and gave a brief overview of the system of codes being developed within the Russian Federal Target Programme (Nuclear Power Technologies for a New Generation for 2010–2015 and for the Future to 2020) for design and safety justification of NPPs with sodium, lead and lead-bismuth coolants, and for the closed nuclear fuel cycle. Simulated designs included fast nuclear reactors with sodium coolant (BN-600, BN-800, BN-1200, MBIR), lead coolant (BREST-OD-300, BN-1200), lead–bismuth coolant (SVBR-100), and other fuel cycle facilities. The code system covers many topics, from the neutron transport in the reactor code to the release of radioactive materials to the environment. The code system provides multi-physics coupled analyses of processes. The status of code development was presented. The system under development allows analysis of both DBA and BDBA and the conduct of PSA analysis.

Z. Zhang (China) delivered a presentation on R&D on simulator of fast reactor in China and provided a brief introduction on the background of simulation with respect to the China Experimental Fast Reactor（CEFR). The full scope real-time simulator was finished by the HEU team in collaboration with the China Institute of Atomic Energy (CIAE) and validated on CEFR. Characteristics of the system, structure and operation of the CEFR, the related research to determine the scope and degree, and formulating models and design systems for the simulation of CEFR had been accomplished. The model and software were developed for 71 of 216 CEFR subsystems, including the reactor physics; primary, secondary and third coolant systems; auxiliary system and passive decay heat removal system, as well as others. The simulator had been applied to undertake the debugging and experimental operation of CEFR and to improve the control methods.

Q. Wu (China) delivered a presentation entitled "Neutronics Experimental Verification for ADS with China Lead based Zero Power Reactor" (CLEAR-0), an accelerator driven system (ADS) project for nuclear waste transmutation, which was launched by Chinese Academy of Sciences (CAS) in 2011. China lead based reactor (CLEAR) was selected as the reference reactor for the CAS ADS. According to the R&D roadmap of CLEAR, a 10 MW(th) lead–bismuth cooled pool-type research reactor named CLEAR-I, coupled with a proton accelerator, will be constructed at the first stage. The test data will be used to validate the calculation method, programme and database used in the nuclear design, and will also support the safety analysis and licence application for CLEAR-I.

N. Mosunova (Russian Federation) presented coupled calculations for fast reactor safety justification with the EUCLID/V1 integrated computer code, the status of the EUCLID/V1 integrated computer code was described. The code was designed for the safety analysis and justification of new generation NPPs deploying liquid metal cooled fast reactors under normal operating conditions, design basis accidents and beyond design basis accidents. The EUCLID/V1 code includes the system thermohydraulics module, the spatial time-dependent neutronics module, the quasi two-dimensional fuel rod module and the module for burnup and decay heat calculations. To validate a coupled modelling of the physical processes in a reactor core and its loops, the experimental data on the BN-600 and BOR-60 transient regimes were used. Full power operation without scram operation (UTOP+ULOF) for the BREST-OD-300 reactor facility and loss of off-site power accident in the BN-1200 reactor had been modelled.

E. Pettersen (Switzerland) presented a paper on simulating circulating fuel fast reactors with the coupled TRACE-PARCS code. System codes represent an alternative to resource intensive computational fluid dynamics methods for solving the thermohydraulic problem of nuclear power reactors characterized by direct power deposition into the combined fuel/coolant, as well as transport of the delayed neutron precursor. Modified neutronics and thermohydraulic coupled TRACE-PARCS code were applied to the Molten Salt Fast Reactor (MSFR) and the capability of the approach to accurately predict its dynamic behaviour had been assessed.

TABLE 6. PRESENTATIONS FROM SESSION 6.6. – COUPLED CALCULATIONS

Chair: **Z. Zhang** and **V. Strizhov**

Id	Presenter	Country	Title
CN245-561 PPT-561	V. Strizhov (Invited)	Russian Federation	Codes of New Generation Developed for Breakthrough Project
CN245-576 PPT-576	Z. Zhang (Invited)	China	Research and Development on Simulator of Fast Reactor in China
CN245-299 PPT-299	Q. Wu	China	Neutronics Experimental Verification for ADS with China Lead-based Zero Power Reactor
CN245-184 PPT-184	N. Mosunova	Russian Federation	Coupled calculations for the fast reactors safety justification with the EUCLID/V1 integrated computer code
CN245-59 PPT-59	E. Pettersen	Switzerland	Simulating circulating-fuel fast reactors with the coupled TRACE-PARCS code

5.6.7. Session 6.7. Experimental Thermal Hydraulics

Four presentations were delivered in Session 6.7. related to experimental studies carried out on fuel assembly thermohydraulic for HLM cooled reactors and liquid metal flow measurement techniques.

J. Kuzina (Russian Federation) presented on heat transfer and temperature non-uniformities in pin bundles with heavy liquid metal coolant with respect to various spacing configurations. The influence of wire wrapping-type and grid-type spacers on the fuel pin surface azimuthal temperature distribution and the heat transfer coefficient were clarified experimentally and analytically under the geometrical conditions of s/d (pin pitch/pin diameter) = 1.28 and 1.33. It was shown that the grid-type spacer is advantageous in the case of free packed fuel pin bundles.

J. Pacio (Germany) reviewed the current status of experimental activities in evaluating thermohydraulic behaviour in the LBE cooled MYRRHA fuel assembly and provided a comprehensive overview of the associated field. This presentation focused on pressure drop, heat transfer, local blockage (heat transfer effect and blockage formation), flow induced vibration and inter-wrapper flow. Uncertainty in the measured heat transfer coefficients was also discussed.

I. Soldatenkov (Russian Federation) discussed depressurization of the heat-exchange tubes that is a supposed failure of normal operation of the steam generator of the lead coolant nuclear reactor in its long-term service. The main object of the work was to evaluate the durability of the heat-exchange tubes based on the results of fretting wear tests. These tests were performed in conditions closed to the real ones, in particular, in respect of identities of the coolant thermal properties, materials and design features of the real steam generator. The experimental data served as the basis for mathematical simulation of fretting wear calculation and evaluation of durability of the heat exchange tubes under the conditions of normal operation.

M. Angelucci (Italy) presented the NACIE-UP facility at ENEA-Brasimone R.C. NACIE-UP is a large scale loop operating with Lead Bismuth Eutectic (LBE) in the range of 180-450°C in free and mixed convection. The facility comprises also a secondary loop in pressurized water with air-cooler to cool the primary LBE. The primary side is instrumented with a prototypical thermal flow meter, a pressure transducer to measure pressure drops across the test section and several thermocouples. The experimental campaigns aimed to study outstanding thermal-hydraulic phenomena such as the heat transfer during transient from forced to natural circulation flow and the flow blockage accident in a fuel assembly. These activities are in support of the front-end engineering design (FEED) of GEN IV ADS prototypes and demonstrators. Some experimental data on heat transfer coefficient obtained in mixed and natural circulation flow regime were also presented.

N. Krauter (Germany) reported on the development of a new eddy current flowmeter (ECFM) and related tests in sodium. The objective of this sensor is its positioning above the fuel subassemblies and the detection of possible blockages of the sodium flow through the multitude of subassemblies. The sensor consists of a number of coils a part of which is fed by an excitation AC current. The assembly of coils is placed in a thimble and the measured flowrate is proportional to the integral flow around this thimble. In the second part of the presentation, the author described local ultrasonic velocity measurements. Here, the objective was to study the flow field resulting from a large electromagnetic pump installed at the PEMDYN facility of CEA. Both measuring techniques were tested at the sodium facility NATAN of HZDR.

TABLE 7. PRESENTATIONS FROM SESSION 6.7. – EXPERIMENTAL THERMAL HYDRAULICS

Chair: **H. Ohshima** and **N. Pribaturin**

Id	Presenter	Country	Title
CN245-340 PPT-340	J. Kuzina	Russian Federation	Heat transfer and temperature non-uniformities in pin bundles with heavy liquid metal coolant at various spacing ways
CN245-283 PPT-283	J. Pacio	Germany	Thermal-hydraulic experiments supporting the MYRRHA fuel assembly
CN245-545 PPT-545 Not presented	I. Soldatenkov	Russian Federation	Testing of the model friction units type of “tube - spacer grid” of the steam generator of the lead coolant nuclear reactor (this presentation was not delivered)
CN245-88 PPT-88	M. Angelucci	Italy	NACIE-UP: a HLM loop facility for natural circulation experiments
CN245-535 PPT-535	N. Krauter	Germany	Eddy current flowrate and local ultrasonic velocity measurements in liquid sodium

5.6.8. Session 6.8. Experimental Facilities

In this session five papers were presented.

G. Gerbeth (Germany) presented paper titled "The DRESDYN project: A new facility for the thermohydraulic studies with liquid sodium". Magentohydrodynamic aspect of moving liquid metal was explained. It was indicated that liquid metal motion creates cosmic magnetic field and using this field, magnetic resonance study is planned. In future, measurements of liquid metal flow in the reactor vessel can be obtained by deploying special sensors. Sodium fire mitigation by liquid argon was also dealt in the presentation. The facility has been erected and experiments will be done in future.

D. Martelli (Italy) delivered presentation on behalf of M. Tarantino on the paper titled "CIRCE-ICE experimental activity in support of LMFR design". The author explained about need of oxygen control to limit the corrosion. It was explained that the oxygen solubility in the lead loop is affected by the temperature of the liquid metal. The experimental facility will be used for qualification of CFD code and decay heat removal system. Development of sensors for oxygen measurement in liquid metal was explained.

F. Serre (France) presented paper titled "PLINIUS-2: a new corium facility and programme to support the safety demonstration of the ASTRID mitigation provisions under severe accident condition". The scenario of severe accident was explained and various provisions in the ASTRID reactor for facilitating mitigation of severe accident were enumerated. It was stated that corium fragmentation study, jet ablation study and qualification of core catcher sacrificial layer study in addition to other experiments related to severe accident are proposed in PLINIUS-2, which is in design stage now and expected to be commissioned in year 2022. The facility is also designed to carry out study related to light water reactor severe accidents.

P. Selvaraj (India) presented the paper titled "Sodium testing of fast reactor components". Qualification tests carried out on PFBR components were explained. The major components which were tested are reactor control system components like control and safety rod (CSR) system and diverse safety rod (DSR) system, fuel handling machine like transfer arm and inclined fuel transfer machine. It was stated that the experiment was carried out which proved heat removal by natural convection method in experimental sodium loop called SADHANA. The demonstration of removal of 355 kW heat by natural convection in sodium loop SADHANA was explained. Experiment to qualify model steam generator of PFBR was also explained.

T. Zhou (China) presented the paper titled "CLEAR-S: A Large Pool-type Components and Thermo-hydraulic Integrated Test Facility for China Lead based reactor". The author explained that the CLEAR-S experimental facility is aimed to carry out experimental study in support of lead based reactor CLEAR, which China had embarked. Thermal hydraulic coupling of fuel assembly study, pool thermal hydraulic study and accident transient study is planned in this experimental facility. Qualification performance test on prototype components will be also carried out in CLEAR-S.

TABLE 8. PRESENTATIONS FROM SESSION 6.8. – EXPERIMENTAL FACILITIES

Chair: **B. K. Nashine** and **V. Semenov**

Id	Presenter	Country	Title
CN245-534 PPT-534	G. Gerbeth (Invited)	Germany	The DRESDYN project: A new facility for thermohydraulic studies with liquid sodium
CN245-89 PPT-89	D. Martelli	Italy	CIRCE-ICE experimental activity in support of LMFR design
CN245-313 PPT-313	F. Serre	France	PLINIUS-2: a new corium facility and programs to support the safety demonstration of the ASTRID mitigation provisions under Severe Accident Conditions
CN245-266 PPT-266	P. Selvaraj (Invited)	India	Sodium testing of fast reactor components
CN245-546 PPT-546 Not presented	V. Sizarev	Russian Federation	On the rational design of fuel assemblies for reactor facilities from the standpoint of providing vibration strength (this presentation was not delivered)
CN245-312 PPT-312	T. Zhou	China	CLEAR-S: A Large Pool-type Components and Thermo-hydraulic Integrated Test Facility for China Lead based reactor

5.6.9. Session 6.9. Research Reactors

This session consisted of five presentations, four from the Russian Federation and one from the Republic of Korea.

D. Euh (Korea) presented the development of flow identification technology for the PGSFR thermal fluidic design validation. The instrumentations to calculate parameters such as, pressure drop using rod internal pressure impulse line, iso-kinetic method for flow rate distribution and split factor and wire mesh system and laser induced fluorescence (LIF) method for mixing factor were successfully developed and applied.

A. Belov (Russian Federation) presented a detailed engineering neutron codes for calculations of fast breeder reactors. Three models with different detailed representation of assemblies in the reactor core were developed. The author explained that to calculate these models' algorithms, codes have been designed. The FUBUKI-1, 2 codes allow to perform the pin-by-pin calculation of the fast breeder reactor core and to keep a history of burnup individually for each pin. The G7 code allows more accurately calculating the assembly with a large width across flats and performing the correct calculation of the fuel assemblies, containing the control rod (the fuel assembly of the BREST-OD-300 reactor).

A. Izhutov (Russian Federation) described the operational experience and experimental capabilities of the BOR-60 reactor. The author summarized that the BOR-60 has been under operation for more than 47 years, the designed lifetime being 20. A decision to extend its lifetime was taken on the basis of good practice of operation, the survey of the state of equipment and materials. Strategic goal of BOR-60 utilization is to provide its operation up to construction and commissioning of the newly experimental fast reactor MBIR at the RIAR site. Thus, the long term research programs launched in BOR-60 will be completed in the MBIR reactor.

D. Klinov (Russian Federation) presented the calculation and experimental analysis of the BN-800 reactor core neutronic parameters at the stage of reaching first criticality followed by rated power testing. The calculated parameter values agree with the measurement results and design values within the declared accuracy of measurement and concluded that no core engineering design adjustment was required.

A. Gulevich (Russian Federation) explained that the experimental fast reactor MBIR is aimed at supporting reactor studies, including testing of new types of fuel and structural materials exposed to various types of coolants to solve the problems of safety, reliability, and economic efficiency. The author described a modified MBIR core that is proposed at the initial stage of MBIR operation, with three irradiation assemblies placed in the central loop channel cells and another irradiation assembly added to flatten power profile. Irradiation volume of the MBIR reactor is increased by four cells (up to 21) while the number of fuel assemblies is decreased from 93 (according to the basic design) to 85 (at the initial stage). The reactor power will be decreased to 137 MW(th) and general rate of damaging dose in MBIR will be 1370 dpa•l/year at the initial stage of reactor operation.

TABLE 9. PRESENTATIONS FROM SESSION 6.9. – RESEARCH REACTOR

Chair: **H. Yamano** and **A. Tuzov**

Id	Presenter	Country	Title
CN245-248 PPT-248	D. Euh (Invited)	Korea, Republic of	Development of Flow Identification Technology for the PGSFR Thermal Fluidic Design Validation
CN245-196 PPT-196	A. Belov	Russian Federation	Detailed engineering neutron codes for calculations of fast breeder reactors
CN245-497 PPT-497	A. Izhutov (Invited)	Russian Federation	BOR-60 reactor operational experience and experimental capabilities
CN245-462 PPT-462	D. Klinov (Invited)	Russian Federation	Calculation and experimental analysis of neutronic parameters of the BN-800 reactor core at the stage of reaching first criticality followed by rated power testing
CN245-438 PPT-438	A. Gulevich (Invited)	Russian Federation	Justification of arrangement, parameters, and irradiation capabilities of the MBIR reactor core at the initial stage of operation

5.6.10. Session 6.10. Other issues of code development and application

Cancelled.

5.6.11. Session 6.11. IAEA Benchmark on EBR-II Shutdown Heat Removal Tests

This session comprised of six presentations.

V. Kriventsev (IAEA), scientific secretary of the FR17 conference, started the session with a presentation describing the overall objectives, scope and accomplishments of the IAEA EBR-II SHRT benchmark. A presentation on behalf of L. Briggs was also done by V. Kriventsev, in which he described more details of the benchmark data and progression.

M. Marchetti (Germany) presented a paper on behalf of B. Vezzoni, giving a detailed comparison of participant results for the SHRT-45R neutronics benchmark. The neutronic benchmark is a subset of the EBR-II benchmark where participants were asked to calculate core neutronic parameters such as core multiplication factor and reactivity feedback effects (radial expansion, coolant expansion, etc.). It was presented that, there were more variations between participants for calculations of the k_{eff} than compared with reactivity feedbacks, which usually agreed within ~15%. A delegate pointed out that it would be useful to see the effect of cross-section uncertainty propagated to these results.

E. Bates (USA) from TerraPower reviewed the SHRT-45R thermal hydraulic, system level modelling results from all participants. Overall, it was shown that participants were consistently good at capturing the pump behaviour and coast-down. However, more deviations were observed for the reactivity feedback contributions and thus fission power during the transient. Overall, radial and coolant expansion were shown to be key reactivity feedback phenomena. Sensitivity studies by various participants showed that heat transfer correlations, axial fuel expansion assumptions, and wire wrap pressure drop correlations tended to have a smaller impact on the results. Listeners enquired about the stable natural circulation flow rate (~5-8 %) in EBR-II and also about the accuracy of modelling pump coast-down.

P. S. Uppala (India) presented a detailed comparison of the sub-channel and CFD models for simulations of the XX09 and XX10 subassemblies in the SHRT-17 and SHRT-45R transients. Overall, it was shown that the CFD was not suitable for the long transient (1000 s) and should be applied for limited time periods. Sub-channel codes demonstrated acceptable accuracy for simulation of the EBR-II subassemblies.

N. Rtishchev (Russian Federation) reviewed the lessons learned by benchmark participants who modelled the SHRT-17 transient. As with SHRT-45R, benchmark participants tended to model the pump coast-down accurately, but results diverged in predicting the steady natural convection flow rate. Predictions of the IHX inlet and outlet temperatures were shown to be very specific to the geometry, and the best results were obtained with codes that could model the specific location of the thermocouples in the IHX.

TABLE 11. PRESENTATIONS FROM SESSION 6.11. – OTHER ISSUES OF CODE DEVELOPMENT AND APPLICATION

Chair: **V. Kriventsev** and **D. Zhang**

Id	Presenter	Country	Title
CN245-361 PPT-361	V. Kriventsev	IAEA	IAEA's Coordinated Research Project on EBR-II Shutdown Heat Removal Tests: An Overview
CN245-4 PPT-4	V. Kriventsev	IAEA	EBR-II Passive Safety Demonstration Tests Benchmark Analyses
CN245-70 PPT-70	M. Marchetti	Germany	IAEA NEUTRONICS BENCHMARK FOR EBR-II SHRT-45R
CN245-372 PPT-372	E. Bates	USA	Conclusions of a Benchmark Study on the EBR-II SHRT-45R Experiment
CN245-118 PPT-118	P. Uppala	India	Thermal Hydraulic Investigation of EBR-II Instrumented Subassemblies during SHRT-17 and SHRT-45R Tests
CN245-84 PPT-84	N. Rtishchev	Russian Federation	Final Results and Lessons Learned from EBR-II SHRT-17 Benchmark Simulations

5.6.12. Track 6. Poster Session

In Track 6, a total of 56 papers were accepted, out of which 46 posters were displayed. There were six posters devoted to the IAEA CRP EBR-II shutdown heat removal tests SHRT-17 and SHRT-45R from China, Germany, India, Japan and the Republic of Korea. As regards SHRT-17, the paper presented by India gave 1D results and the paper by Japan 1D and 3D coupled code results. Four posters were devoted to SHRT-45R, where thermohydraulic parameters were compared with measured data, and feedback reactivity coefficients were compared against other code results. The results compare reasonably well, despite the complex geometry and process. There were 15 posters related to CFD and thermohydraulic studies on various aspects of fast reactors. Studies related to nuclear data uncertainties were the subject of six posters. Three posters were related to calculation of neutronic parameters for BOR-60 under various conditions. Neutronic calculations were also presented in 17 posters.

5.7. TRACK 7 – FAST REACTORS AND FUEL CYCLES: ECONOMICS, DEPLOYMENT AND PROLIFERATION ISSUES

5.7.1. Session 7.1. Sustainability of Fast Reactors

Session 7.1 comprised of four presentations, three from the Russian Federation, and one from Japan. The presentations briefly described prospects of fast reactors development and tools for their assessment from the point of sustainability.

S. Maeda (Japan) presented "Current Status of Next Generation Fast Reactor Core & Fuel Design and Related R&D in Japan". The presentation showed that the next generation fast reactor being investigated in Japan, are aiming at several targets from areas such as safety, environmental, and economics.

A. Andrianov (Russian Federation) presented "Performance and sustainability assessment of nuclear energy deployment scenarios with fast reactors: advanced tools and application". The paper presented the toolkit developed in the National Research Nuclear University MEPhI for a performance and sustainability assessment of nuclear energy deployment scenarios with fast reactors providing a solution to the problem of optimizing and comparing nuclear energy deployment scenarios with fast reactors in multiple criteria formulation. Some results of implementation of this toolkit were presented.

A. Yegorov (Russian Federation) presented "Comparison of Innovative Nuclear Energy Systems Based on Selected Key Indicators and Their Weighing Factors". The paper presented a methodological study on comparison of nuclear energy systems whose commissioning and commercial-scale operations are in the planning stage. The key indicators from the study affect not only the assessment of reactor facility, but also the characteristics of nuclear energy system as a whole.

E. Marova (Russian Federation) presented "Evaluation results of BN-1200 compliance with the requirements of Gen IV and INPRO". The paper presented the results of preliminary assessment of the BN-1200 project in terms of safety and economics. The assessment showed that BN-1200 meets level of safety and economical characteristics in comparison to the BN-800 reference unit and ensures sustainable development of the nuclear energy system.

TABLE 1. PRESENTATIONS FROM SESSION 7.1. – SUSTAINABILITY OF FAST REACTORS

Chair: **A. Gulevich** and **S. Maeda**

Id	Presenter	Country	Title
CN245-269 PPT-269	S. Maeda (Invited)	Japan	Current Status of Next Generation Fast Reactor Core & Fuel Design and Related R&D in Japan
CN245-194 PPT-194	V. Usanov	Russian Federation	Assessment of a nuclear energy system based on the integral indicator of sustainable development
CN245-7 PPT-7	A. Andrianov	Russian Federation	Performance and sustainability assessment of nuclear energy deployment scenarios with fast reactors: advanced tools and application
CN245-434 PPT-434	A. Yegorov	Russian Federation	Comparison of Innovative Nuclear Energy Systems Based on Selected Key Indicators and Their Weighing Factors
CN245-399 PPT-399	E. Marova	Russian Federation	Evaluation results of BN-1200 compliance with the requirements of Gen IV and INPRO

5.7.2. Session 7.2. Economics of Fast Reactors

The topics discussed at the session included ensuring the competitiveness of nuclear energy in the long term perspective; comparison of the competitiveness of NPP projects with fast, thermal reactors and alternative generation; market and economic analysis of nuclear cogeneration by NPPs with fast reactors; economic analysis of fuel cost components at NPPs with VVER and BN-type reactors; competitiveness of fast reactors when working in the system alongside LWRs and approaches to evaluation of non-standardized equipment for fast reactors.

D. Tolstoukhov (Russian Federation) discussed on the competitiveness of nuclear energy. It was noted that modern NPP projects with thermal reactors almost exhausted reserves with regards to improving competitiveness and cannot guarantee the long-term effectiveness of nuclear energy. Now, the Russian Federation has embarked on the PRORYV project, which is directed at the development of competitive technologies using a closed nuclear fuel cycle. Within the framework of the PRORYV project, the requirements identified with respect to the technical and economic parameters allow the long-term effectiveness of nuclear energy to be ensured.

M. Frignani (France), presented on fast reactors and cogeneration, low temperature applications (<200°C) typically include district heating or desalination (e.g. ALLEGRO GFR project). Medium temperature applications (~450–500°C) are the focus of current R&D (e.g. ASTRID SFR and ALFRED LFR projects). High temperature applications (>750°C) are assessed over a long-term perspective (e.g. LFR/GFR projects). Cogeneration has a good synergy with small to medium sized reactors, owing to flexibility of operating multiple units of small size and the improved safety features which may be realized in small power reactors.

V. Dekusar (Russian Federation) showed that analytical calculations of the levelized unit fuel cost (LUFC) for BN-1200 with MOX fuel (closed NFC) was comparable to VVER-TOI and uranium fuel (open NFC). It was explained that to reduce the LUFC for the BN-1200, it is first necessary to reduce the costs of fabrication of MOX fuel. However, the initial data have some uncertainty attached and the results are to be clarified at some point in the future. The potential synergy between SFRs (breeder) and LWRs (100% MOX fuel loading) provides a number of advantages from an economic perspective. This is due to the fact that the plutonium produced by breeders can be valued, at best, by feeding an LWR at a smaller investment cost than a FBR. It can also make the FBR competitive earlier. The LWR with a high conversion ratio will be useful in developing the SFR market.

G. Mathonnière (France) explained that synergy between SFRs and LWRs provides a number of advantages from an economic perspective and it is highly probable that future nuclear fleets that no longer burn natural uranium will be composed of both breeder SFRs and high conversion ratio LWRs using 100% MOX fuels. Using SFRs in such a fleet improves their economic competitiveness.

N. Molokanov (Russian Federation), presented that the feasibility of creating a new generation of fast neutron reactors with heavy liquid metal coolants is largely governed by the development of new structural materials. At the same time, the task of estimating the cost of new equipment is complicated by the absence of direct analogues. The practice of estimating the value of equipment under design shows the need for using all the existing tools (analogy based, resource based (bottom-up) estimation methods). The estimation should be conducted, taking into account the design stage, the applicability factors (completeness of data and availability of information), labour input and permissible error.

TABLE 2. PRESENTATIONS FROM SESSION 7.2. – ECONOMICS OF FAST REACTORS

Chair: **D. Tolstoukhov** and **G. Mathonnière**

Id	Presenter	Country	Title
CN245-82 PPT-82	M. Frignani	France	Fast Reactors and Nuclear Cogeneration: A Market and Economic Analysis
CN245-98 PPT-98	D. Tolstoukhov	Russian Federation	Providing the competitiveness of nuclear energy in the implementation of PRORYV project
CN245-435 PPT-435	V. Dekusar	Russian Federation	Comparative analysis of electricity generation fuel cost component at NPPs with WWER and BN-type reactor facilities
CN245-296 PPT-296	G. Mathonniere	France	How to take into account the fleet composition in order to evaluate Fast Breeder Competitiveness
CN245-536 PPT-536	N. Molokanov	Russian Federation	Equipment cost estimation for pilot demonstration lead cooled fast-neutron reactor BREST-OD-300

5.7.3. Session 7.3. Non-Proliferation Aspects of Fast Reactors

Session 7.3. comprised of five presentations, two from Italy, two from the Russian Federation and one from France.

M. Frignani (Italy), presented the status and perspectives of industrial supply chain for fast reactors and explained that the identified challenges of the EU nuclear industry with respect to fast reactors are mainly related to maintaining the current supply chain capabilities, defining specifications of critical components, developing new materials and fabrication and inspection techniques, ensuring the necessary accreditation and quality. Key aspects and requirements applicable to the supply chain were discussed and were mainly related to the implementation of requirements for fast reactors into the nuclear codes and standards, the extension of quality requirements, and certifications. It was identified that a challenge for fast reactor development in the long term is to minimize or avoid code/country-related barriers. The main critical components of fast reactor concepts have characteristics and requirements that might represent a challenge for their design, materials, manufacture and constructability for the industry, thereby requiring further investments on R&D and qualification. This will represent a business opportunity for the industry. A wide variety of companies in the European nuclear might be able to supply most of the identified components, based on their current capabilities. The technical specifications for critical components are not readily available and further analysis will be needed.

L. Volpe (France), presented the paper "ASTRID - An Original and Efficient Project Organization". CEA is the contracting authority and industrial architect of ASTRID Project, an industrial prototype of Gen IV sodium fast reactor. This reactor of 600 MW(e) is integrating French and international SFRs feedback, especially in domains of safety, operability and ultimate wastes transmutation. Mr Volpe explained that the objectives of ASTRID require the implementation of innovating engineering and management methods which go beyond the current feedbacks; and significantly different compared to the former projects. This presentation described the industrial set-up implemented through the series of collaboration agreements with the different project partners, thereby making it possible to take into account the interests of the industrial parties at a very early stage and to integrate recent feedback from other projects. Close coordination between the prototype construction project and the R&D support programs was emphasized and stated that it will also contribute to the success of the project. CEA as project owner is managing the ASTRID project in a collaborative approach based on a series of concrete organizational methods.

O. Saraev (Russian Federation), summarized concepts on closing a nuclear fuel cycle in a two-component system with thermal and fast neutron reactors. The Russian nuclear power industry already possesses a two-component system with installed capacity of power units with thermal neutron reactors at 25,640 MW(e), and the installed capacity of power units with fast neutron reactors with 1,485 MW(e). Considering the planned new power units, both these components are most likely to exist side by side throughout the rest of the 21st century and both the components have their advantages and disadvantages. The presentation discussed the advantages and disadvantages with a postulated benefit of closing the fuel cycle.

M. Frignani (Italy), presented on FALCON advancements towards the implementation of the ALFRED Project. Innovative nuclear energy systems, with breakthrough concepts belonging to a new generation of nuclear technologies (Gen IV), have the potential to meet the highest performances in terms of sustainability, safety, proliferation resistance and economics. The vision for a near future benefitting of new lead cooled fast reactors (LFR) as clean, resource effective, safe and economic, hence sustainable, innovation-intensive power sources has been the driving force for the pan-European efforts which eventually led to the ALFRED Project. Lately, the implementation of European deployment strategies of Gen IV technologies is being postponed towards the 2050s. Although a longer-term perspective may lead to a reduced industrial interest, the intrinsic and passive safety features of the LFR design make it an optimum candidate for the SMR segment. The ALFRED reactor is being revised to meet the goal of a feasibly deployable lead cooled SMR concept, based on technologically-ready solutions and compatible with the short term global needs related to de-carbonization and security of energy sources. FALCON is considered the proper incubator for the concept development and will be a pole of attraction for partners interested in the heavy liquid metal technology.

A. Chebeskov (Russian Federation) presented the “GIF Proliferation Resistance and Physical Protection (PR&PP) Evaluation Methodology: Status, Applications and Outlook”. The PRPPWG (PR&PP working group) has developed an evaluation methodology that likely represents the most comprehensive publicly available PR&PP tool that can inform the design process of any nuclear technology. It was presented that the PR&PP methodology is aligned with international efforts to improve the effectiveness and efficiency of safeguards. It represents an enabling tool for “safeguards by design”, and, in conjunction with the risk and safety working group of GIF, a natural manifestation of the integration of the previously noted safety, security, and safeguards (sometimes called “3S”) linkage within the culture of nuclear technology design. It is expected that the PRPPWG will continue to work with the SSCs to implement pilot applications of the PR&PP methodology, as well as maintain cognizance of international developments and engagement with other groups within the international non-proliferation community. The PR&PP methodology will be maintained as necessary to retain its relevance and applicability to the development of new and emerging nuclear systems, primarily within GIF but also for the broader nuclear community.

TABLE 3. PRESENTATIONS FROM SESSION 7.3 – NON-PROLIFERATION ASPECTS OF FAST REACTORS

Chair: **A. Chebeskov** and **S. Kim**

Id	Presenter	Country	Title
CN245-486 PPT-486	M. Frignani	Italy	Status and perspectives of industrial supply chain for Fast Reactors
CN245-294 PPT-294	L. Volpe	France	ASTRID - An original and efficient project organization
CN245-309 PPT-309	O. Saraev	Russian Federation	Closing up nuclear fuel cycle in a two component system with thermal and fast neutron reactors
CN245-485 PPT-485	M. Frignani	Italy	FALCON advancements towards the implementation of the ALFRED Project
CN245-526 PPT-526	A. Chebeskov	Russian Federation	The GIF Proliferation Resistance and Physical Protection (PR&PP) Evaluation Methodology: Status, Applications and Outlook

5.7.4. Session 7.4. Fuel Cycle Analysis

Session 7.4. included papers on fast reactors fuel cycle analysis, including non-proliferation issues; burning of transuranic elements; and cost evaluation and nuclear power scenarios for PWR–FR, matching development with the closed nuclear fuel cycle.

The fast reactors represent a promising technology with the potential to ensure the sustainable development of nuclear energy. It can tap the inexhaustible energy locked into natural uranium and thorium and burn the long-lived radioactive waste at the same time. Combinations of thermal and fast reactors can work well together, and the recycling of actinides from the thermal systems into fast systems can reduce the worldwide actinide inventory.

A. Rineiski (Germany) summarized a study on partitioning and transmutation (P&T) for nuclear waste management, which was conducted in Germany for possible fuel cycle options and systems for burning/utilization of transuranic elements in the German and European frameworks. Mr Rineiski explained that for P&T, several ADS/FR options are available, with the SFR being the most mature one. SFRs proposed in Europe can accommodate different P&T options, but their design may have to be modified to incorporate TRUs and maintain acceptable safety parameters.

M. Kim (Republic of Korea) presented the analysis of various thorium fuel options for the SFR undertaken in the Republic of Korea. In the analysis, the TRU transmutation performance was tested for various design options with thorium fuel loaded in the core of an SFR. The analysis was conducted on three fuel type categories: (i) oxide fuel, (ii) metal fuel and (iii) nitride fuel. It was concluded that as a potential nuclear fuel, thorium improves nuclear transmutation, safety and nuclear proliferation resistance in SFRs.

K. Zhou (China) presented the nuclear energy development scenario base with U–Pu multi-cycling with PWR, FR and CNFC in China. It was explained that owing to limitations of uranium resources, China is evaluating the development of the FR and the closed nuclear fuel cycle. The paper presented analyses of four nuclear power scenarios for PWR–FR, matching development with the closed nuclear fuel cycle. It was concluded that to achieve faster development of nuclear power capacity, it is necessary to have sufficient natural uranium to support the large-scale development of PWRs, and as a result, to accumulate sufficient plutonium from spent fuel reprocessing to load the FR core, which is a prerequisite for the rapid development of FRs. The large-scale development of FRs requires sufficient reprocessing capacity.

V. Artisyuk (Russian Federation) discussed the topic of small fast modular reactors by using the example of the Russian fast SMR (SVBR-100) with chemically inert coolant based on LBE that can reduce the risk of severe accidents. The author explained that the SVBR-100 demonstrates key characteristics in the fields of safety and reliability, non-proliferation technology support, opportunities to operate in the different nuclear fuel cycles, and safe SNF and radioactive waste management. The analysis of the SVBR-100 fuel cycle together with reprocessed uranium involvement showed a decrease in the fissile attractiveness of fresh and irradiated fuel in terms of ATTR (advanced nuclear fuel cycle assessment) methodology, as well as the savings in natural uranium consumption amounting to ~10% over the reactor's lifetime.

G. Toshinsky (Russian Federation) presented an analysis of the developed design documentation for first of kind nuclear power plant that shows needs of some design optimization to create a competitive serial nuclear power plant based on the SVBR-100 reactor. The analysis defined the main directions and scales for this optimization, which are reduction in equipment and construction costs, reduction of specific indicators (nuclear power plant site area, volume of the nuclear island buildings and mass of a heat-mechanic equipment to installed capacity), decrease in the number of personnel and increase in the installed capacity of the reactor.

A. Chebeskov (Russian Federation) discussed on the problem of non-proliferation. It was stressed that it is very important to follow IAEA safeguards at an early stage in the development of new nuclear reactor designs and appropriate technologies of nuclear fuel cycle — the so-called 'safeguards by design'.

TABLE 4. PRESENTATIONS FROM SESSION 7.4. – FUEL CYCLE ANALYSIS

Chair: **A. Khaperskaya** and **X. Huo**

Id	Presenter	Country	Title
CN245-487 PPT-487	A. Rineiski	Germany	Fast reactor systems in the German P&T and related studies
CN245-242 PPT-242	M. Kim (Invited)	Korea, Republic of	Performance Analysis of Various Thorium Fuel Options for the Sodium Cooled Fast Reactor
CN245-412 PPT-412	K. Zhou	China	Primary Analysis on The Nuclear Energy Development Scenario base on the U-Pu Multicycling with PWR, FR and CNFC in China
CN245-343 PPT-343	V. Artisyuk	Russian Federation	Analysis of the SVBR-100 nuclear fuel cycle by means of the advanced nuclear fuel cycle assessment methodology (ATTR)
CN245-90 PPT-90	G. Toshinsky	Russian Federation	SVBR Project: status and possible development
CN245-104 PPT-104	A. Chebeskov	Russian Federation	Fast Neutron Reactors, Fuel Cycles and Problem of Nuclear Non-Proliferation

5.7.5. Track 7. Poster Session

Nine papers from the representatives of the Russian Federation, Germany, the UK, and China were presented at the poster session of Track 7 “Fast Reactors and Fuel Cycles: Economics, Deployment and Proliferation Issues”.

The main subject of the posters was the analysis of the current level of development of the closed nuclear fuel cycle (NFC) with fast reactors (FR) and its prospects in terms of the ability to address the current challenges of sustainable development of nuclear energy (economic competitiveness, operational safety, non-proliferation etc.).

The posters analyzed a wide range of fuel compositions for irradiation in fast reactors (fresh fuel from enriched uranium, fuel from weapons grade plutonium, fuel from regenerated light water reactor fuel) with an analysis of its behaviour in the core, characteristics and strategies for spent nuclear fuel management.

A number of posters devoted to the development of methods and computer programs for modelling scenarios for the development of nuclear power in the transition to fast neutron reactors were presented.

The conclusions common to all the posters are:

- Confirmation of physical realizability of the closed nuclear fuel cycle by fast neutron reactors;
- The existence of issues related to the production of plutonium in fast neutron reactors (in blankets), which require additional analysis in terms of non-proliferation;
- A sufficiently high stage of readiness for the pilot implementation of elements of the nuclear power system on the basis of fast reactors and a good potential for their compliance with the basic principles and requirements of the IAEA, INPRO, and Generation IV International Forum.

5.8. TRACK 8 – PROFESSIONAL DEVELOPMENT AND KNOWLEDGE MANAGEMENT

5.8.1. Session 8.1. Professional Development and Knowledge Management – I

Session 8.1. was devoted to professional development and knowledge management. The section was focused on HRD to support the development of fast reactor programmes and associated fuel cycles as well as international cooperation. There were five presentations in total.

Two national presentations were given, including one on the development and deployment of the knowledge management portal for fast breeder reactors in India (V. Arasappan) and one on topical issues for the training of specialists for fast nuclear power engineering and the closed nuclear fuel cycle in the Russian Federation (G. Tikhomirov).

Three presentations focused on international cooperation within EURATOM (R. Garbil, EC), within Gen IV (K. Mikityuk, Switzerland) and within the ALFRED project (S. Bortot, Sweden).

V. Arasappan (India) of Indira Gandhi Centre for Atomic Research presented an overview of the activities ongoing in the Indira Gandhi Centre for Atomic Research on the development and deployment of the taxonomy based knowledge management portal for fast breeder reactors, which currently stores 5,300 reports and 20,000 drawings related to all the stages in the lifecycle of fast reactors being operated and being constructed in India. The portal is an asset of the IGCAR, without access permission from outside.

G. Tikhomirov (Russian Federation) of MEPhI emphasized the importance of educational support of the innovative projects. The motto 'new projects require new university programmes' is realized in MEPhI where the new department was established to support ROSATOM's innovative 'PRORYV' project. This department, in cooperation with the nuclear industry and R&D institutions, implements MSc and PhD programmes by focusing on development of SFR, LFR and associated closed fuel cycles.

R. Garbil (EC) presented the experience gained, acting instruments and ongoing programmes in the European Union in terms of research, energy industry, and education with special emphasis given to European fission R&D work programmes on partitioning and transmutation, as well as the SCN Academy which focuses on supporting the MYRRHA, ASTRID and Jules Horowitz reactor projects.

K. Mikityuk (Switzerland) presented recent activities provided by a special E&T taskforce established under the umbrella of the Gen IV project in 2016. In 2016–2017, the taskforce organized series of webinars on topical issues of fast reactor development and closed fuel cycles. Clear instructions on how to gain access to archived materials were given and webinars scheduled for 2017–2018 were advertised.

S. Bortot (Sweden) provided an overview of the European ARCADIA project (Advanced Lead Cooled Fast Reactor, European Demonstration) and its implementation in Romania. The most challenging issues were identified. Among them were material science, I&C, HLM chemistry and thermohydraulics.

TABLE 1. PRESENTATIONS FROM SESSION 8.1. – PROFESSIONAL DEVELOPMENT AND PROLIFERATION ISSUES

Chair: **V. Artisyuk** and **T. Ivanova**

Id	Presenter	Country	Title
CN245-382 PPT-382	V. Arasappan	India	Development and Deployment of Knowledge Management Portal for Fast Breeder Reactors
CN245-533 PPT-533	G. Tikhomirov (Invited)	Russian Federation	Topical issues of training of specialists for fast nuclear power engineering and the closed nuclear fuel cycle
CN245-351 PPT-351	R. Garbil	EC	'EURATOM success stories' in facilitating pan-European E&T collaborative efforts
CN245-9 PPT-9	K. Mikityuk	Switzerland	Gen IV Education and Training Initiative via Public Webinars
CN245-427 PPT-427	S. Bortot	Sweden	A proposal for a pan-European E&T programme supporting the development and deployment of ALFRED

5.8.2. Session 8.2. Professional Development and Knowledge Management – II

Session 8.2. comprised four contributions addressing IAEA initiatives in the field of knowledge management, as well as discussion of the Russian Federation's PRORYV project.

C. Batra (IAEA) presented on the IAEA's fast reactor knowledge portals (FRKP) and catalogues. The FRKP represents a collaborative effort to preserve fast reactor data and knowledge. Mr Batra highlighted the main objectives of the FRKP, which are related to the storage of information related to FRs, and to the preservation and accessibility of existing data and information on FRs. The FRKP hosts publicly available material, as well as restricted and classified information. Moreover, the FRKP is also designed as a platform to host data related to new developments. The FRKP is structured to include document repositories, project workspaces for the IAEA's coordinated research projects (CRPs), technical meetings (TMs) and forums for discussion, etc. In the portal, a taxonomy based search tool is implemented, which helps in the use of new semantic search capabilities for improved conceptual retrieval of documents. Finally, interest in the FRKP has been reconfirmed at several TMs held in the field of FRs and the related fuel cycle, along with constant support from the IAEA Technical Working Group on Fast Reactors (TWG-FR).

V. Kriventsev (IAEA) presented an overview of IAEA activities in his paper on "FR Technology Development: Current State and Future Vision". He first introduced the IAEA's FR technology Development team and its objectives, which are related to: the creation of a platform for exchange of information; the production of technical reports on FRs and ADS; the support of R&D activities; the identification of common international safety approaches, design criteria and guidelines; the sharing of data on experimental facilities; the development, verification and validation of advanced simulation tools through experimental benchmarking; the implementation of opportunities for education and training; and the collection and preservation of existing documents, data and information. After describing the programme, as well as the composition and activities of the TWG-FR, which is the driving force in this area, the author then discussed the programmatic areas and related CRPs: (i) modelling and simulations; (ii) technical support; (iii) FR safety; (iv) education and training and (v) knowledge preservation. For each area, Mr Kriventsev gave examples of activities performed, and concluded his presentation by showing the list of workshops and conferences organized and in particular summarized the participation in terms of papers and attendees at the FR17 conference.

M. Noskov (Russian Federation) discussed personnel training for the PRORYV Project at the Seversk Technological Institute of NRNU MEPhI. The author discussed the implementation plan of the pilot project PRORYV which is based on the creation of a new generation of fast reactor technologies and the closed nuclear fuel cycle. The plan foresees the creation of a research and demonstration power complex (RDPC), including the BREST-OD-300 reactor, and a nuclear fuel fabrication plant. Moreover, an integral part of this project concerns the training and professional development of staff who will operate the complex. The training addresses educational levels which range from high school to university, MSc, PhD and post-doc. The programme foresees career guidance to young students in order to stimulate their interest in natural sciences, engineering, etc. Furthermore, university programmes and PhD activities are also included in the PRORYV project and highly qualified scientific professionals from NRNU MEPhI and ROSATOM are involved in the teaching. Finally, the author explained that the educational programme makes use of modern interactive and multimedia training technologies and also provides hands-on training on nuclear facilities.

O. Azpitarte (Argentina) presented details of the "IAEA NAPRO Coordinated Research Project: Physical Properties of Sodium — Overview of the Reference Database and Preliminary Analysis Results". The author explained that the objective of the NAPRO CRP is to collect, assess and disseminate a comprehensive dataset of sodium properties to support SFR research, design, analysis and development. Mr Azpitarte discussed the status of the CRP, which will have as an output the publication of a handbook on the thermophysical properties of sodium. He showed some examples of collected data and their analysis in order to assess their consistency and to indicate the most reliable dataset. He noted that despite the fact that sodium properties are commonly considered to be 'established', inconsistencies and gaps were identified both for the modelling, and for the experimental database. Key issues in data collection were identified, included the missing information on the details concerning measurement methods and data uncertainty, as well as the purity of sodium samples and their handling. In conclusion, the CRP work has indicated that the body of experimental work undertaken on sodium and liquid metals in general is considerable. However, the quality of some datasets could require revision and may not be suitable for direct use in nuclear facility design and operation.

TABLE 2. PRESENTATIONS FROM SESSION 8.2. – PROFESSIONAL DEVELOPMENT AND KNOWLEDGE MANAGEMENT – II

Chair: **G. Tikhomirov** and **C. Fazio**

Id	Presenter	Country	Title
CN245-31 PPT-31	C. Batra	IAEA	IAEA's Fast Reactors Knowledge Portals and Catalogues
CN245-359 PPT-359	V. Kriventsev	IAEA	Overview of the IAEA Activities in the Field of Fast Reactor Technology Development: Current State and Future Vision
CN245-538 PPT-538	M. Noskov	Russian Federation	Personnel training for the "PRORYV" project at the Seversk Technological Institute of NRNU MePhI
CN245-132 PPT-132	O. Azpitarte	Argentina	IAEA NAPRO Coordinated Research Project: Physical Properties of Sodium Overview of the Reference Database and Preliminary Analysis Results

5.8.3. Track 8. Poster Session

In track 8, a total of seven posters were selected, however only two were presented. One from the Russian Federation on comparative analysis of nuclear energy lexicon and another one from the International Atomic Energy Agency (IAEA) on overview of the international cooperation and collaboration activities initiated and performed under the Technical Working Group on Fast Reactors in the last 50 years.

6. SUMMARY OF PANEL SESSIONS

6.1. PANEL I – DEVELOPMENT AND STANDARDIZATION OF SAFETY DESIGN CRITERIA (SDC) AND SAFETY DESIGN GUIDELINES (SDG) FOR SODIUM COOLED FAST REACTORS (SFR)

Sodium cooled fast reactor (SFR) is the most mature design in establishing safety design criteria among six reactor conceptual designs selected by Generation IV International Forum (GIF). For the global harmonization on safety, GIF assigned a dedicated task force on SFR safety. Safety design criteria for SFRs were presented and discussed in FR13 conference in Paris. At FR17 in Yekaterinburg, six experts from several GIF countries and India shared their opinions and debated on safety design guidelines: R. Nakai (Japan), Y. Okano (Japan), P. Gauthe (France), S.C. Chetal (India), J. Yoo (Republic of Korea), and Y. Ashyrko (Russian Federation). The panel was moderated by V. Kriventsev (IAEA).

The panel discussion started with a brief introduction of the actual state and visions presented by all six participants who gave an update on SDC and SDG for SFRs since the FR13 conference in Paris. The first two presentations by R. Nakai and Y. Okano were delivered on behalf of GIF.

R. Nakai introduced **"**The Safety Design Guideline Development for Gen IV SFR Systems" developed by GIF task force that has been developing a set of safety design guidelines (SDG) to support practical application of safety design criteria (SDC) for GIF SFR systems. The main objective of the SDG development is to assist SFR developers and vendors to utilize the SDC in their design process for improving the safety in specific topical areas including the use of inherent/passive safety features and the design measures for prevention and mitigation of severe accidents. The first report on "Safety Approach SDGs" aimed to provide guidance on safety approaches covering specific safety issues on fast reactor core reactivity and on loss of heat removal. The report was drafted and disseminated to international organization/group such as the IAEA and GSAR for the external review. The second report on "SDGs on key Structures, Systems and Components (SSCs)" is under development and focuses on the functional requirements for SSCs importance to safety; reactor core system, reactor coolant system, and containment system.

Y. Okano informed that GIF task force completed development of safety design criteria (SDC) for the Gen IV SFR systems in May 2013. SDC reflect high level GIF safety and reliability goals (excellence in operational safety and reliability, and reduced likelihood and degree of core damage) and follows GIF basic safety approach (application of defence-in-depth and emphasis on inherent and passive safety features so that safety is built-in to the design, not added-on). The SDC reflects SFR's common and superior passive and inherent safety features on shutdown and cooling. The SDC report aimed to establish reference criteria for safety design of structures, systems and components and achieve harmonization of safety approaches among GIF member states. Following its public release, SDC report was distributed to international organizations and national regulatory bodies for review and feedback. The feedback addressed several subjects such as sodium chemical reactivity, security consideration with safety impact, minor actinide bearing fuel, severe accident consideration on containment, independence of Defence-in-Depth (DiD) on support systems, glossary/expressions on quality assurance, inherent power reduction with complementary shutdown, Gen IV system safety level, and situations to be practically eliminated. GIF resolutions will be summarized in a separate report along with the revised version of SDC report.

GIF presentations were followed by national presentations from France, India, Republic of Korea and the Russian Federation.

P. Gauthe pointed out that from a general perspective, Gen IV reactors should excel in safety and, as part of a continuous improvement process, provide safety enhancements. Safety objectives for Gen III reactors are already very ambitious and, therefore, relevant for Gen IV reactors. These Gen III objectives deal with severe accident prevention, severe accident mitigation, which is considered in the frame of the fourth defence in depth level and response to external hazards, including natural hazards of extreme intensity. The safety objectives for the Gen IV SFRs are similar to those of the most recent French Gen III reactor projects, with especially the mitigation of the core melt accident, with the objective of very limited releases such that no off-site measures are necessary. If measures are nevertheless necessary (e.g., restrictions on consumption of a crop), they shall be limited in time and space with sufficient time for their implementation. Even temporary evacuation of populations should not be necessary and only sheltering, limited in time and space, shall be envisaged. Considering the already very ambitious Gen III objectives as the reference, Gen IV reactors will excel in safety with improved safety design and more robust safety demonstration."

S. C. Chetal briefed on the safety criteria for future SFRs in India. He stated that basic approach for safety criteria will be based on the feedback from licensing process of PFBR, safety criteria issued by regulatory body for thermal reactors and important safety criteria requirements envisaged for Gen IV SFRs. The enhanced safety requirements for future SFRs include limit on sodium void coefficient, in-vessel primary sodium purification, additional decay heat removal system from primary sodium pool, passive secondary shut down system and third shut down system, additional line of support for core support structure, and tighter requirements for design against external events.

J. Yoo explained compliance of Korean SFR safety design approaches with Gen IV safety design criteria (SDC). The design criteria of PGSFR are based on Korean regulation of general design criteria. The GIF SDC has been implemented into the PGSFR principal design criteria in preliminary safety information document. Amendment of Nuclear Safety Act in Korea came into effect since 2016. Change in regulatory environment includes accident management plan for all new reactors.

Y. Ashyrko discussed SFR safety requirements and approaches and their correspondence to Gen IV SFR safety design criteria. Main Russian standards and regulations applied for SFR safety justification were enumerated. Basic approaches to SFR safety analysis set in these regulations were described and compared with safety design criteria developed within the GIF. The correspondence of the Russian regulations to the GIF safety design criteria was demonstrated. Some specific requirements of the Russian norms and rules to SFR safety were described. The evolution of main safety characteristics of the Russian SFRs was shown in the example of BN-600, BN-800 and BN-1200. In particular, key approaches and design decisions on the provision of BN-1200 safety were explained. Analysis of these safety approaches and technical decisions implemented in the BN-1200 design and the achievement of its safety characteristics demonstrated their compliance with safety design criteria for Gen IV SFR.

The panel discussion continued from the question on the practical elimination of the need for the off-site emergency response. Participants presented their views on the principle of 'practical elimination' (which is one of the GIF corner-stone objectives) and agreed that a clear explanation for the public would be useful. Participants also agreed that robust demonstration to eliminate the need for off-site emergency response is required. The future steps in developing SDG/SCD were described and discussed.

6.2. PANEL II – SMALL AND MEDIUM SIZED FAST REACTORS

The IAEA Department of Nuclear Energy devotes a number of its initiatives to support the development and deployment of small and medium sized fast reactors. This initiative is opted due to the strong conviction of their potential advantages for enhancing nuclear energy supply and security among the Member States that are embarking in the new ventures or expanding their existing facilities. Y. Kim (Republic of Korea), G. Toshinsky (Russian Federation), G. Grasso (Italy), J. Krepel (Switzerland) and S. Qvist (Sweden) discussed the potential benefits and supported the small and medium sized fast reactors. The panel was moderated by S. Monti (IAEA).

With a brief introduction by S. Monti on the actual state and visions of small and medium sized fast reactors, the panel discussion started based on the presentations by all five participants. The progress made since the FR13 conference in Paris was also intensely debated.

Y. Kim (Republic of Korea) explained several potential advantages of these facilities. They include, much simplified system design, higher level of safety, low power density with long core lifetime and possibility of no refuelling over whole lifetime, higher fuel burnup, higher thermal efficiency, closed fuel cycle with extremely proliferation-resistant fuel recycling, etc. Although above mentioned advantages are highly beneficial, they also pose specific challenges. These challenges can be on various aspects such as economy, licensing framework, operation and maintenance costs, fuel supply, safeguard issues and physical protection, etc.

G. Toshinsky (Russian Federation) discussed about SVBR-100 as a possible option for developing countries. He debated on when choosing the NPP for developing countries; a number of requirements regarding the type of the NPP proposed for use should be taken into account. These include an assured elimination of the severe accident when the NPP is used in a co-generation mode, the export-availability of the NPP, and a simplified operation of the NPP considering the requirements to the personnel's qualification etc.

G. Grasso (Italy) presented the work on the "Core of the LFR-AS-200: Robustness for Safety". He emphasized that the concept of LFR-AS-200 has been conceived to provide a credible option for an innovative SMR integrating the safety and sustainability performances that are proper of LFRs with the economic competitiveness, that is required to compete in this emerging market segment. The design of the core of the LFR-AS-200 was also presented, introducing the rationales and discussing the key performances that can be expected once in operation. The system is able to achieve 10% of fuel utilization, delivering 200 MW(e) for 5 cycles, each 16-month long. These results have shown that the system poses no threat not only to the people and environment but also to the protection of the investment.

J. Krepel (Switzerland) discussed the eligibility of Small Molten Salt Fast Reactor (S-MSFR). The author mentioned that current commercial nuclear reactors faced two major problems. Their public acceptance is influenced by the fear of severe accidents, the production of nuclear waste and the capital cost for construction due to the growing safety requirement. Both, the capital cost and the risk of severe accidents may be reduced by the so called small and medium sized or modular reactors (SMRs). Molten Salt Reactor (MSR) as one of the Gen IV system can also be designed as SMR. Nonetheless, the inherent features of MSR with liquid fuel can provide similar advantages even at high nominal powers.

S. Qvist (Sweden) talked about the development of the autonomous reactivity control (ARC) system to ensure inherent safety of fast reactors while having a minimal impact on reactor performance and economic viability. He summarized, the state-of-the-art of these development efforts and the results of full transient analysis of ARC-equipped fast reactor cores. The ARC system is in active development at the University of California Berkeley & Argonne National Laboratory in the US and at Uppsala University in Sweden.

7. SUMMARY OF YOUNG GENERATION EVENT

7.1. PANEL III – YOUNG GENERATION EVENT

In order to support the future envisioned in the UN 2030 Agenda for Sustainable Development, the IAEA organized a Young Generation Event (YGE) as a part of the FR17 Conference. The YGE event was divided into two parts; (i) “Young Innovator Challenge”, and (ii) “Young Global Leader Challenge”. The Young Innovator Challenge called participants to submit a research proposal related to one or more of the conference’s eight thematic tracks. Applicants of the Young Global Leader challenge were asked to prepare a speech and an associated presentation on the topic of ‘Next Generation Nuclear Systems for Sustainable Development’. Five scientists under the age 35 were selected as winners of the Young Innovator Challenge, which solicited original research proposals on Fast Reactors (FRs) or innovative nuclear technologies capable of contributing to the United Nations Sustainable Development Goals related to energy and climate change. In addition, one winner was chosen in the second part of the contest - the Young Global Leader Challenge.

This section provides the summary of the winning research proposals.

1. Innovative cold trap filtration technologies for reliable and economic exploitation of lead-bismuth eutectic cooled systems

The twofold purpose of the research was defined. The first part consisted of the development of a cold trap technology specifically for LBE. This included the selection principle, materials and conditions of such a cold trap filtration system. The second part focused on the engineering implementation and operation of the cold trap in an actual reactor design. It was also highlighted that it is of prime importance to develop an efficient, robust and economically performing cold trap filtration system for LBE cooled systems. The eventual goal of the research is to accelerate the international advancement in LBE and lead coolant technology not only for large power reactors but also for small, modular and medium sized reactors with various applications.

Accurate control of LBE chemistry is an important issue for the design of reliable LBE cooled systems. Corrosion of structural steels leads to an increase of impurity levels in the LBE over time. These impurities could accumulate and cause blockages in heat exchangers or the fuel assembly of a nuclear system. Safe operation of LBE cooled systems, such as the Gen IV nuclear reactors require the control of impurity levels in the liquid coolant. The research aimed at resolving this.

2. Stability and bifurcation analysis of sodium boiling in a Gen IV SFR reactor core

The aim of this research project was to strengthen the existing knowledge on sodium boiling phenomenology by developing an innovative methodology with respect to the nuclear safety context, based on stability and bifurcation analysis through a semi-analytical procedure. As an output, a more reliable understanding of boiling phenomenology in the reactor core could be achieved. Sodium boiling is a very dynamic two-phase flow phenomenon, and numerical tools such as system codes are challenging to set and validate for transients which lead to sodium boiling.

3. Development of Reverse Flow Blockage Device for Primary Sodium Pumps of Fast Breeder Reactor

The main goal of this research project was to conceptualize and develop an innovative reverse flow blockage device. Fast Breeder Reactors (FBRs) generally have two or more Primary Sodium Pumps (PSP) operating in parallel. However, the PSPs of many FBRs are not designed with Non-Return Valves (NRV) citing various demerits like impact on valve stuck open or stuck close on the safety of the reactor, reduced impeller submergence for the pump, enhanced pressure drop in the circuit, etc. Hence, in the event of a PSP trip, the reactor has to be shut down and the flow from the other operating pumps will be passing through the reactor core and the tripped PSP. In the view of attaining better economics and flexibility in plant operation, an active mechanism concept called as “sleeve valve mechanism” was presented. This mechanism could block the flow reversal through a tripped PSP. Such a mechanism could also perform the function of NRV without having the demerits of NRV. Thus, the mechanism is aimed at reducing the PSP size and power requirements (as it doesn’t cause additional pressure drop under normal operating conditions) and enabling the reactor operation at enhanced power levels even after a pump trip.

4. A flexible, easy-to-use-easy-to-obtain computational tool to stimulate the development of innovative reactor concepts

The aim of the research project presented was to develop and test an easy-to-use, open-source computational tool which is able to resolve the complex nature of advanced nuclear reactors. The tool is intended to be able to couple neutronics and thermalhydraulics, natively support two-phase fluids, model a wide range of innovative fuels and coolants in specialized geometries, and allow simple propagation of results for further processing (e.g. thermal-mechanical calculations). It should do so while providing a modular and easily-extendable code structure, intuitive and simple ways to model reactor systems, acceptable computational requirements, and compatibility with parallel processing. For many researchers, computer simulations represent the most viable route for developing an innovative concept towards large-scale adoption. An easily obtainable and simple to use computational tool could, therefore, stimulate the development of innovative nuclear reactor concepts.

5. Development of tri-isoamyl phospahte (TiAP) based solvent extraction process as an alternate method for the processing of metallic alloy fuels (U-Pu-Zr and U-Zr)

The research project aimed at studying and developing an aqueous based reprocessing method based on the well-established PUREX process as an alternate method to pyro-chemical reprocessing of metallic fuels. In the PUREX process, dissolution of metallic fuels in the nitric acid medium is an important step. The feed solutions generated after the dissolution could be directly employed for the subsequent solvent extraction cycles using PUREX process. Tri-n-Butyl Phosphate (TBP) in n-alkane diluent medium (typically 1.1M of extractant in n-dodecane) has been utilized as a versatile solvent for various separation processes in nuclear technology. However, the experience gained in the last six decades has brought out a few drawbacks of TBP that are of concern during the reprocessing of Fast Reactor fuels. The major drawbacks of TBP include third phase formation with tetravalent metal ions e.g. plutonium (IV), higher aqueous solubility, radiation degradation etc.

The objective of the research is to study the dissolution aspects of metallic alloy fuels in nitric acid media with or without addition of fluoride ions and generate feed solutions containing uranium, plutonium and zirconium, which can be employed in subsequent reprocessing steps, to investigate the explosive nature of metallic alloys during dissolution in nitric acid medium and to demonstrate the continuous counter-current mixer-settler runs with U-Pu-Zr feed solution using Tri-isoamyl Phosphate (TiAP) based solvent and comparison with TBP.

The winner of the "Young Global Leader Challenge" delivered a speech on "**How the next generation of people will shape the next generation of nuclear".** This speech reflected the importance of nuclear energy in the future energy mix and emphasized the contributions that could be made in order to move closer to these goals. It also suggested different ways, through which the young generation could be motivated to reinforce the nuclear knowledge for the future, such as knowledge preservation, openness to new ideas and continuous communication and exchange of information. Ideas related to how nuclear energy could address some of the Sustainable Development Goals were also discussed, mainly focusing on goals 7 (affordable and clean energy), 9 (industry, innovation and infrastructure) and 13 (climate action). The speech was a youthful exuberance of enthusiasm and commitment of the young generation to make a difference, developing nuclear as an effective tool for addressing climate change, reducing the carbon footprint of a growing planet and contributing to sustainable development.

List of YGE challenge winner:

'Young Innovator' challenge

Rank	First Name	Last name	Country	Title of research proposal
1	Kristof	Gladinez	Belgium	Innovative cold trap filtration technologies for reliable and economical exploitation of lead-bismuth eutectic cooled systems
2	Edouard	Bissen	France	Stability and bifurcation analysis of sodium boiling in a Gen IV SFR reactor core
3	S	Aravindan	India	Development of Reverse Flow Blockage Device for Primary Sodium Pumps of Fast Breeder Reactor
4	Eirik	Pettersen	Switzerland	Developing an open-source multi-physics tool for simulating advanced nuclear reactors
5	Balija	Sreenivasulu	India	Development of Tri-iso-Amyl Phosphate (TiAP) based solvent extraction process as an alternate method for the processing of metallic alloy fuels (U-Pu-Zr and UZr)

'Young Global Leader' challenge

First Name	Last name	Organization	Title of Speech
Luke	Lebel	France	How the Next Generation of People will shape the Next Generation of Nuclear

7.2. YOUNG GENERATION EVENT WORKSHOP

Under the topic "Filling the gap: Training Young Generation" a workshop was organized by the joint efforts of the IAEA (International Atomic Energy Agency) and the IYNC (International Youth Nuclear Congress).

Knowledge transfer between senior and junior professionals is an important matter, even more so for the Fast Reactor community. There is no doubt that there is a gap that needs to be bridged, and knowledge that has to be shared and transferred between the experienced engineer and the young university graduate.

During the first part of the workshop several presentations were given by senior and young experts from different member states. They allowed the participants to gain a better understanding of the current status quo in the Fast Reactors community. CEA, JAEA and ROSATOM, as well as the IAEA and IPPE gave interesting insights and shared their ideas and suggestions for knowledge management and transition.

The second part of the workshop encouraged its participants to contribute to the important issue of knowledge transfer and share their own ideas and suggestions on this topic. Two working groups were formed, both consisting of senior and young professionals. Lively discussions took place while exchanging opinions on topics like bridging the gap between senior and junior professionals, academia and industry, tools that can be used to manage and preserve knowledge in a more efficient way and increasing the interest in the next generation of young nuclear engineers.

At the end of the workshop each group presented their outcomes, agreeing that is important to involve young people already in school, and that there should be an easier access to information for children, teenagers and students. Encouraging young girls and showing them that there is also a place for them in the Fast Reactor industry, plays an important role.

8. CLOSING SESSION

8.1. CHAIR OF THE INTERNATIONAL ADVISORY COMMITTEE

Hideki Kamide

Chair, International Advisory Committee, Closing Summary

With more than 400 technical contributions, the international conference of fast reactors and related fuel cycles (FR17) attracted participation from all over the world. Overall, a total of 47 oral technical sessions with 243 presentations, three plenary sessions with eleven presentations and two poster sessions with 206 presentations were organized. The table below provides the titles of each technical track and plenary session.

TECHNICAL AND PLENARY SESSIONS

Plenary Session	**National and international fast reactor programmes**
Track 1.	Innovative Fast Reactor Designs
Track 2.	Fast Reactor Operation and Decommissioning
Track 3.	Fast Reactor Safety
Track 4.	Fuel Cycle Sustainability, Environmental Considerations and Waste Management Issues
Track 5.	Fast Reactor Materials (Fuels and Structures) and Technology
Track 6.	Test Reactors, Experiments, Modelling and Simulations
Track 7.	Fast Reactors and Fuel Cycles: Economics, Deployment and Proliferation Issues
Track 8.	Professional Development and Knowledge Management

Three Plenary sessions were organized for three days each morning with keynote speeches from major fast reactor technology development countries and few international organizations: Russian Federation (two), China, France, India, Japan, Republic of Korea, EC/JRC, Generation IV International Forum (GIF), OECD/NEA, and IAEA. During the plenary session the keynote speakers discussed the national and international fast reactor programmes, providing the fast reactor community with most up-to-date information regarding the development of fast reactor programmes.

In Track 1 on innovative fast reactor designs, out of 55 accepted papers, 38 were selected for oral presentation and 17 for poster presentation. The track was divided into eight technical sessions.

TABLE 1. TRACK 1. INNOVATIVE FAST REACTOR DESIGNS. TECHNICAL SESSIONS

Session	**Title**
1.1	SFR Design and Development – I
1.2	SFR Design and Development – II
1.3	System Design and Validation
1.4	Core and Design Features – I
1.5	LFR Design & Development
1.6	Core and Design Features – II
1.7	ADS and Other Reactor Designs
1.8	Innovative Reactor Designs

Various presentations encouraged technical information exchange and discussion related to experience and lessons learned from reactors currently in operation or being commissioned, design of future SFRs in various Member States including other reactor types, i.e. LFR, MSR, Burning Wave reactor, ADS, GCR, safety design criteria for SFRs, SFR systems and components research, design, development, qualification and validation, alternative options for power conversion system, core design features and characteristics of core with respect to other fuels such as metal and nitride, small size reactors enabling specific applications etc. One of the major observations was that the designs of future SFRs, which are currently under development are focusing on a high level of safety, achieved through provision of advanced safety features and economic design.

In Track 2 on fast reactor operation and decommissioning, out of 27 accepted papers, 15 were selected for oral presentation and 12 for poster presentation. The track was divided into three technical sessions:

TABLE 2. TRACK 2. FAST REACTOR OPERATION AND DECOMMISSIONING. TECHNICAL SESSIONS

Session	Title
2.1	Commissioning and Operating Experience of Fast Reactors – I
2.2	Commissioning and Operating Experience of Fast Reactors – II
2.3	Decommissioning of Fast Reactors and Waste Management

The presentations provided useful discourse on the operational experience and it was observed that some reactors, still in operation have given fruitful feedback over more than 35 years of operation (FBTR, BN-600). Also, reactors in the commissioning stage or which have been recently started have successfully achieved the performance objective, which promotes confidence in the future operation of the plant (BN-800, PFBR). The lessons learnt and operational experience feedback are integrated into the design of future reactors (ASTRID). Decommissioning of fast reactors is successful, without major technical issues, and the treatment of decommissioning waste has reached the industrial stage (Superphenix, other SFR prototypes or research reactors)

In Track 3 on fast reactor safety, out of 70 accepted papers, 35 were selected for oral presentation and 33 for poster presentation. The track was divided into seven technical sessions:

TABLE 3. TRACK 3. FAST REACTOR SAFETY. TECHNICAL SESSIONS

Session	Title
3.1	Innovative Reactor Designs
3.2	Core Disruptive Accident
3.3	Probabilistic Safety Assessment
3.4	Sodium leak/fire and other safety issues
3.5	General Safety Approach
3.6	Safety Analysis
3.7	Core Disruptive Accident Prevention

In the technical sessions pertaining to this topic, great attention was paid to systematic consideration of severe accidents in SFR and LFRs. Discussions were carried out related to the analysis of Passive Safety Systems (PSS) for reactor shutdown and decay heat removal under severe accident conditions. The importance of Safety Design Criteria and Guidelines were recognized as a cornerstone of the LFR safety and the significance of prevention and mitigation of CDAs was recognized.

In Track 4 on fuel cycle sustainability, environmental considerations and waste management issues, out of 31 accepted papers, 16 were selected for oral presentation and 15 for poster presentation. The track was divided into three technical sessions:

TABLE 4. TRACK 4. FUEL CYCLE SUSTAINABILITY, ENVIRONMENTAL CONSIDERATIONS AND WASTE MANAGEMENT ISSUES. TECHNICAL SESSIONS

Session	Title
4.1	Fuel Cycle Overview
4.2	Reprocessing and Partitioning
4.3	Partitioning and Sustainability

It was observed that major efforts are being put on developing efficient and innovative recycling processes for dealing with high burnup, high plutonium content, fast neutron reactor fuels (oxides but also nitrides or carbides) by hydro, pyro or a combination of pyro and hydro technologies. There is also significant work being done in the field of minor actinide recovery and separation for transmutation technologies for plutonium multi-recycling with limited involvement. The principal efforts are aimed at decreasing nuclear cycle impact on environment and improving economic efficiency of fuel reprocessing technologies. Countries have different approaches to R&D in spent fuel reprocessing, which depends on fuel type selected for short and long term perspectives. Poster session showed a moving trend from natural experiment to the modelling and simulations.

In Track 5 on fast reactor materials (fuels and structures) and technology, out of 106 accepted papers, 47 were selected for oral presentation and 55 for poster presentation. The track was divided into 10 technical sessions:

TABLE 5. TRACK 5. FAST REACTOR MATERIALS AND TECHNOLOGY. TECHNICAL SESSIONS

Session	Title
5.1	Advanced Fast Reactor Fuel Development – I
5.2	Advanced Fast Reactor Fuel Development – II
5.3	Advanced Fast Reactor Cladding Development - I
5.4	Advanced Fast Reactor Cladding Development - II
5.5	Large Component Technology – I
5.6	Liquid Metal Technologies
5.7	Chemistry Related Technology
5.8	Structural Materials
5.9	Large Component Technology – II
5.10	Fuel modelling and Simulation

In this track, considerable new information was presented. For different preferences of fuel type for FBRs, such as metallic, MOX, mixed carbides and mixed nitrides, participants had their own reasons for justifying their choice which was fully discussed. The similar situation concerns the choice of liquid metal coolant (sodium, lead and lead-bismuth) and invited equal amount of constructive discourse. The most important discussions were devoted to results of fuel and cladding qualification under irradiation and post-irradiation examination results. It was observed that significant progress has taken place in the field of fuel modelling and simulation, in particular the modelling of fuel composition behaviour, cladding materials and structures.

In Track 6 on Test Reactors, Experiments, Modelling and Simulations, out of 117 accepted papers, 53 were selected for oral presentation and 57 for poster presentation. The track was divided into eleven technical sessions:

TABLE 6. TRACK 6. TEST REACTORS, EXPERIMENTS, MODELLING AND SIMULATIONS. TECHNICAL SESSIONS

Session	Title
6.1	CFD and 3D Modelling
6.2	Thermal Hydraulics Calculations and Experiments
6.3	Neutronics – I
6.4	Neutronics – II
6.5	Uncertainty Analysis and Tools
6.6	Coupled Calculations
6.7	Experimental Thermal Hydraulics
6.8	Experimental Facilities
6.9	Research Reactors
6.10	Other issues of code development and application
6.11	IAEA Benchmark on EBR-II Shutdown Heat Removal Tests

This track was clearly the biggest track of the conference in terms of number of papers submitted and accepted and it can be easily seen that numerical simulations are extensively used in support of the fast reactor technology. Accurate description of complicated phenomena in LMFRs requires massive calculations in 3D. Coolant properties are vital in obtaining reliable data for thermalhydraulic calculations. It was observed that modern codes are capable to simulate complex phenomena with acceptable accuracy. A special session on "IAEA Benchmark on EBR-II Shutdown Heat Removal Tests" was also organized as a technical session under this track. This session highlighted the results and achievements of the IAEA's coordinated research project (CRP) on the same topic. Various CRP participants and the IAEA presented the work done under the CRP.

In Track 7 on Fast reactors and Fuel Cycles: Economics, Deployment and Proliferation issues, out of 29 accepted papers, 21 were selected for oral presentation and 7 for poster presentation. The track was divided into four technical session:

TABLE 7. TRACK 7. FAST REACTORS AND FUEL CYCLES: ECONOMICS, DEPLOYMENT AND PROLIFERATION ISSUES. TECHNICAL SESSIONS

Session	Title
7.1	Sustainability of Fast Reactors
7.2	Economics of Fast Reactors
7.3	Non- Proliferation Aspects of Fast Reactors
7.4	Fuel Cycle Analysis

Fuel cycle plays an important role in assuring the sustainability of nuclear power and fast reactor aim to close the fuel cycle. The discussion related to cogeneration pointed out its capability for good synergy with SMRs owing to flexibility of multiple small-size units and improved safety features. It was observed that the feasibility of the new generation of innovative fast reactors with heavy liquid metal depends on development of new structural materials. Large scale development of fast reactors requires sufficient reprocessing capacity and the importance of the IAEA Safeguards by Design was highlighted.

In Track 8 on Professional development and knowledge management, out of 16 accepted papers, nine were selected for oral presentation and seven for poster presentation. The track was divided into two technical sessions:

TABLE 8. TRACK 8. PROFESSIONAL DEVELOPMENT AND KNOWLEDGE MANAGEMENT. TECHNICAL SESSIONS

Session	Title
8.1	Professional Development and Knowledge Management - I
8.2	Professional Development and Knowledge Management - II

Human resource development is much needed to support the advancement of fast reactor technologies and associated fuel cycle as well as international cooperation. Development of national and international portals (with data, documents etc.) for fast reactors, that includes past experiences will assist in retaining and transferring of knowledge. The task force on E&T activities initiated under the umbrella of Gen IV is currently producing webinars for knowledge dissemination. Several projects and initiatives promote Knowledge Transfer and Management as well as E&T opportunities. These projects and initiatives include: ARCADIA, ALFRED, PRORYV, IAEA Fast Reactor Portal, etc. It is expected, based on the recommendations of specialists, that in future it will be necessary to develop educational tools to use FR international and national databases, portals, etc. and transfer knowledge and expertise to young generation.

Conclusion

Based on the discussions through various technical sessions, plenary sessions, workshops etc., it can be concluded that the fast reactor technology remains a proven option as a sustainable source of energy for many generations to come. The international cooperation on technology of fast reactors and related fuel cycles is vital. The sodium cooled fast reactor technology remains the most mature technology and now the focus is on enhancing safety and improving economic efficiency. The fast reactor community also agreed that it is of common interest and benefit to continue this series of conference every four years under the aegis of the International Atomic Energy Agency.

8.2. CONFERENCE GENERAL CO-CHAIR

Closing speech as provided, verbatim.

Mikhail Chudakov

DDG-NE, Conference General Co-Chair, Closing Remarks

Dear distinguished delegates, ladies and gentlemen,

I would like to express my gratitude for your participation at the International Conference on Fast Reactors and Related Fuel Cycles: Next Generation Nuclear Systems for Sustainable Development, organized by the IAEA and hosted by the Government of the Russian Federation through ROSATOM.

Thanks to the assistance of the Member States, which you represent, and thanks to your personal efforts, dear members of the group, the IAEA organizes international conferences on fast reactors and related fuel cycles every four years.

Yekaterinburg is undoubtedly the best site today to host this Conference as it provides an opportunity to visit the world's largest operating sodium cooled fast reactor. I would like to thank the Russian Federation and the Russian Federation's State Atomic Energy Corporation "ROSATOM" for providing all the necessary support. I really appreciate that.

Having the honour to be the final speaker of the Conference, I would like to provide you with some final statistics that I think are worth mentioning.

We have an outstanding level of participation, totalling almost 600 participants, drawn from 27 countries and six international organizations. This is strong confirmation that there is a robust and growing interest in fast reactor and related fuel cycle technology.

This is also reflected in the number of scientific contributions that we received during the past year in preparation for this event. We were delighted to have received 449 technical papers from 27 countries, of which 243 were presented orally and 206 presented as posters. In addition, we received eleven keynote speeches and eleven presentations delivered at two technical panels, as well as six contributions at the Young Generation Event Panel. These six presentations were delivered at the YGE workshop this morning.

In three plenary sessions, we had the opportunity to learn about the latest status of development of fast reactors and related fuel cycle technology in countries with the active fast reactor programmes. Four international organizations, including the IAEA, presented their members' visions on the topic (total eleven keynotes).

Monday:

Russian Federation: E. Adamov from the Russian Federation presented the results obtained during the 5 years of the PRORYV Project, which confirmed the technological feasibility of its fundamental principles and made it possible to proceed to the practical development stage and to a new nuclear technological platform.

China: H. Yu from China Institute of Atomic Energy gave an overview of the nuclear energy programme in China, which is developing nuclear energy with a capacity about 30 GW(e), with 35 reactors in operation and another 19 under construction, with the target to reach 58 GW(e) by 2020 and about 400-500 GW(e) by 2050, although the average utilization of nuclear power in China has declined for three years. Several new generation nuclear energy systems are currently under study in China with the focus on sustainable nuclear fuel cycles to meet the future demands.

CEA: S. Pivet from CEA presented the status of the French fast reactor programme. The nuclear energy will remain one of the pillars of the future French low carbon energy mix. The closed fuel cycle associated with fast neutron reactors will lead to drastic improvement in uranium resources management and important reduction in footprint and radiotoxicity of final wastes.

Tuesday:

India: A. K. Bhaduri, Director of Indira Gandhi Centre for Atomic Research, summarized the details of the Indian fast reactor programme and discussed its status and R&D achievements. The FBR programme in India has several aspects. The construction of the Fast Breeder Test Reactor (FBTR) afforded comprehensive experience in construction and operation and in the provision of material irradiation data, including for reactor and energy conversion systems. The Prototype Fast Breeder Reactor (PFBR) is intended for technical and economic demonstration of the system.

Japan: Y. Sagayama (Japan Atomic Energy Agency, JAEA) announced that Japan’s Fourth Strategic Energy Plan was approved by the Cabinet in April 2014. Japan will continue to promote the nuclear fuel cycle in terms of the efficient use of resources and will carry out R&D for fast reactor commercialization, taking advantage of international cooperation.

Republic of Korea: J. Yoo of the Korea Atomic Energy Research Institute delivered a presentation on the status of the SFR development programme in the Republic of Korea.

Russian Federation (concluding, as a host country): Mr. Tuzov, Director of RIAR, delivered a presentation on the Russian Federation’s research and pilot fast reactors, which are considered as the basis for the development of commercial reactor technologies.

Wednesday (International Organizations)

European Commission: In the European Commission, contributions to the development of fast reactor systems are based on platforms, initiatives and alliances created to distribute R&D resources.

Generation IV International Forum (GIF): F. Gauché introduced the Generation IV International Forum (GIF) and gave details of the recent GIF activities over the past four years. New members have been included in GIF since FR13 in Paris. A detailed description of six Gen IV reactor designs was presented, compared and discussed. SFRs remain in focus to GIF members and also attract the interest of the private sector.

Organization for Economic Co-operation and Development: T Ivanova from the OECD Nuclear Energy Agency (NEA) highlighted the support that NEA continues to provide in the fundamental science and technology that underpin fast reactors, serving as a forum for the exchange of information and for promoting collaborative activities.

IAEA: The presentation given by J. Phillips covered the principal activities of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) in the areas of nuclear energy system sustainability assessment and in whole system scenario analysis in support of long term planning for sustainable development of nuclear energy. The particular focus of this presentation was on projects that directly involve fast reactors, related fuel cycles and the potential for international cooperation.

The scientific contributions summarized by H. Kamide in his concluding report on the technical sessions have all exhibited significant technical expertise and proved the vitality and level of innovation in the fast reactor field and in related fuel cycle activities.

The past week has offered us very exciting opportunities to explore the development of fast reactors and related fuel cycles.

I would like to express my thanks to V. Pershukov for his support as General Chair of the Conference and for his efforts during the opening session, and to S. C. Chetal for acting as Honorary Chair of the Conference.

It is also my pleasure to thank our International Advisory Committee, especially H. Kamide, who served as chairperson during the preparations for the Conference.

Of course, I also want to highlight the enormous effort made by the International Scientific Programme Committee, which, under the Chair L. Bolshov, evaluated over 550 abstracts and then reviewed and finally accepted 449 technical papers (243 oral and 206 posters).

I also want to thank the Moderators of the Panel Sessions, V. Kriventsev and S. Monti, the Chair of the Young Generation Workshop, E. Adamov, the Chair of the Young Generation Panel, C. Xerri, and Moderator of the Young Generation Panel, C. Batra.

Please let me also thank all Track Leaders of our 8 Technical Tracks and, of course, the Chairpersons of the Technical Sessions.

It is also worth mentioning the continuation of the Young Generation Event, which sends a very strong message that the fast reactor and related fuel cycle field is a technology for the future.

In conclusion, it is my great pleasure to convey special thanks to all our very committed staff members from ROSATOM who worked diligently during the past year to make this event a great success.

I also want to thank our scientific secretaries, A. González Espartero and V. Kriventsev, along with their associates A. Lazykina and C. Batra, as well as M. Neuhold and V. Jordanosvka, who have all worked determinedly to make this Conference a very successful one.

I wish you good and safe travel to your home countries and wish you all the best in your activities and declare this International Conference on Fast Reactors and Related Fuel Cycles in Yekaterinburg is closed.

ANNEX I

STATISTICAL DATA

I-1. GENERAL INFORMATION

Organized by the:	IAEA (NENP/NEFW)
Hosted by the:	Government of the Russian Federation
Through the	State Atomic Energy Corporation “ROSATOM”
Location:	Yekaterinburg EXPO
Total no. of participants and observers:	558
No. of participants from Member States:	509
No. of participants from developed countries:	164
No. of participants from developing countries:	345
No. of participants from organizations:	18 (including IAEA)
No. of countries:	27
No. of organizations represented: (including IAEA)	6
No. of presentations:	449 technical papers 6 opening statements 11 plenary session presentations 11 panel presentations 6 YGE presentations 6 closing statements 243 oral technical presentations
No. of posters:	206 poster presentations
Scientific Secretaries:	Amparo González-Espartero, NEFW Vladimir Kriventsev, NENP
Scientific Support:	Chirayu Batra, NENP Anastasia Lazykina, NEFW
Conference Co-ordinators:	Martina Neuhold, MTCD Viktorija Jordanovska, MTCD

I-2. PARTICIPANTS DATA

Total Participants and Observers: 558

Participants from Member States: 509

Argentina	2	Kazakhstan	1
Belgium	5	Luxembourg	2
Brazil	1	Mexico	4
Belarus	3	Mongolia	1
China	26	Poland	1
Czech Republic	3	Korea, Republic of	34
Egypt	1	Russian Federation	244
Estonia	1	Slovenia	1
France	78	Slovakia	5
Germany	10	Sweden	1
Hungary	5	Switzerland	5
India	15	United Kingdom	1
Italy	10	United States of America	9
Japan	40		

Participants from Organizations: 18

EU (European Union, for EC and EC-JRC)	4
IYNC (International Youth Nuclear Congress)	1
Generation IV International Forum (GIF) (also designated by France)	2
ENEN (European Nuclear Education Network Association)	1
IAEA (International Atomic Energy Agency)	8
OECD/NEA (Nuclear Energy Agency of the Org. for Economic Co-operation and Development)	2

ANNEX II

LIST OF POSTERS

For ease of reference the Id numbers have been hyper-linked to the conference website in the online publication. The full papers and presentations are also included on the CD-ROM attached to this publication for easy access offline.

Poster Session I

Track 1

Posters	Id	Presenter	Country	Title
P1-01	CN245-1	Z. Gholamzadeh	Iran	Computational investigation of nuclear waste incineration efficiency in a subcritical molten salt driven by 50-100 MeV protons
P1-02	CN245-17	K. Yoon	Korea, Republic of	Mechanical Design Evaluation of Fuel Assembly for PGSFR
P1-03	CN245-22	J. Sienicki	USA	Advanced Energy Conversion for Sodium Cooled Fast Reactors
P1-04	CN245-155	N. Kim	Korea, Republic of	High temperature design and evaluation of forced draft sodium-to-air heat exchanger in PGSFR
P1-05	CN245-162	C. Park	Korea, Republic of	Structural Design and Evaluation of a Steam Generator in PGSFR
P1-06	CN245-185	G. Grasso	Italy	The core of the LFR-AS-200: robustness for safety
P1-07	CN245-226	M. Belonogov	Russian Federation	The optimization of core characteristics of fast molten salt reactor based on neutron-physical and thermal-hydraulic calculations and the analysis of fuel cycle closure options
P1-08	CN245-235	I. Volkov	Russian Federation	The lead cooled fast reactor transition to equilibrium operating conditions
P1-09	CN245-268	C. Kim	Korea, Republic of	Neutronic Self-sustainability of a Breed-and-Burn Fast Reactor Using Super-Simple Fuel Recycling
P1-10	CN245-276	X. Chen	Switzerland	Possibility studies of a boiling water cooled traveling wave reactor
P1-11	CN245-302	Y. Kotov	Russian Federation	Application of Heterogeneous Fuel Assemblies in the Core of Modular Fast Sodium Reactor
P1-12	CN245-371	S. Bogetic	USA	3-D Core Design of the TRU-Incinerating Thorium RBWR Using Accident Tolerant Cladding
P1-13	CN245-388	B. Hombourger	Switzerland	On the feasibility of Breed-and-Burn fuel cycles in Molten Salt Reactors
P1-14	CN245-394	A. Lizin	Russian Federation	Selection of carrier salt for molten salt fast reactor
P1-15	CN245-400	F. Chanteclair	France	ASTRID reactor: design overview and main innovative options for Basic Design

Track 1

Posters	Id	Presenter	Country	Title
P1-16	CN245-436	A. Sorokin	Russian Federation	Investigations in a substantiation of high-temperature nuclear energy technology with fast-neutron reactor cooled by sodium for manufacture of hydrogen and other innovative applications
P1-17	CN245-478	M.Vanderhaegen	Belgium	Fast Reactors - The Belgian Regulatory Approach
P1-18	CN245-499	Y. Osheyko	Russian Federation	The concept of 50-300 MW(e) modular-transportable nuclear power plant with sodium coolant and a gas turbine

Track 3

Poster	Id	Presenter	Country	Title
P1-19	CN245-6	T. Ishizu	Japan	Model validation of the ASTERIA-FBR code related to core expansion phase based on THINA experimental results
P1-20	CN245-25	D. Grabaskas	USA	A Mechanistic Source Term Calculation for a Metal Fuel Sodium Fast Reactor
P1-21	CN245-26	D. Grabaskas	USA	Advanced Reactor PSA Methodologies for System Reliability Analysis and Source Term Assessment
P1-22	CN245-55	M. Bucknor	USA	An Assessment of Fission Product Scrubbing in Sodium Pools Following a Core Damage Event in a Sodium Cooled Fast Reactor
P1-23	CN245-85	F. Wang	China	Study on the limits of confinement leakage rates of pool-type sodium cooled fast reactor
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P1-26	CN245-150	J. Lüley	Slovakia	Assessment of the reactivity effects of Gas cooled Fast Reactor
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P1-28	CN245-177	D. Lemasson	France	Benchmark Between EDF And IPPE On The Behaviour Of Low Sodium Void Reactivity Effect Sodium Fast Reactor During An Unprotected Loss Of Flow Accident
P1-29	CN245-192	I. Shvetsov	Russian Federation	Decay-heat removal in accidents in fast reactors with liquid metal coolant
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P1-32	CN245-205	S. Qvist	Sweden	Passive Shutdown Systems for Liquid Metal cooled Fast Reactors
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P1-34	CN245-225	S. Poumerouly	France	Impact of the irradiation of an ASTRID-type core during an ULOF with SIMMER-III
P1-35	CN245-233	D. Blagodatskykh	Russian Federation	ROUZ code: CFD approach for assessment of radiation situation during atmosphere radioactivity releases within an industrial site
P1-36	CN245-324	N. Girault	France	Main outcomes from the JASMIN project: development of ASTEC-Na for severe accident simulation in sodium cooled fast reactors
P1-37	CN245-327	S. Raghupathy	India	Design and Development of Stroke Limiting Device for Control & Safety Rod Drive Mechanisms (CSRDMs) of future FBRs
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P1-39	CN245-336	V. Karuppanna	India	Numerical and Experimental Investigations of Tube-to-Tube Interaction of Air Heat Exchangers of PFBR under Seismic Excitations
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P1-41	CN245-356	M. Jeltsov	Sweden	Voiding of ELSY primary system during steam generator leakage
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P1-45	CN245-433	I. Mickus	Sweden	Preliminary transient analyses of SEALER
P1-46	CN245-450	K. Mikityuk	Switzer-land	ESFR-SMART: new Horizon-2020 project on SFR safety
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P1-49	CN245-530	S. Beils	France	CFD Simulation of Corium / Materials Interaction for Severe Accidents

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P1-54	CN245-35	V. Smolenski	Russian Federation	Thermodynamics and separation factor of lanthanides and actinides in system “liquid metal-molten salt”
P1-55	CN245-147	I. Makeyeva	Russian Federation	modelling technologies of fuel cycles
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P1-58	CN245-260	A. Salyulev	Russian Federation	Electrical conductivity of molten LICL-KCL eutectic with components of spent nuclear fuel
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P1-61	CN245-311	A. Glazov	Russian Federation	On-site nuclear fuel cycle of “BREST” reactors
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P1-70	CN245-49	I. Portnykh	Russian Federation	Investigation of Radiation-Induced Swelling of EK-164 Steel, an Advanced Material for BN-600 and BN-800 Claddings
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P2-45	CN245-209	Š. Čerba	Slovakia	Actual Status of the Development of Multigroup XS Libraries for the Gas cooled Fast Reactor in Slovakia
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P2-54	CN245-280	Q. Zhao	China	the simulation of reactor physics for China Experimental Fast Reactor
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P2-64	CN245-529	J-M. Hamy	France	Advanced Coupling Methodology for Thermal-hydraulic calculations
P2-65	CN245-532	V. Yufereva K. Kalugina	Russian Federation	Methodical uncertainty of criticality precise calculations for fast lead reactor
P2-66	CN245-551	A. Tutukin	Russian Federation	Application of CFD simulation to validate the BRES-OD-300 primary circuit design
P2-67	CN245-552	K. Sergeenko	Russian Federation	LES-simulation of heat transfer in a turbulent pipe
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P2-69	CN245-604	I. Hwang	Korea, Republic of	Characterization of LBE Non-isothermal Natural Circulation by Experiments with HELIOS Test Loop and Numerical Analyses

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P2-73	CN245-39	D. Wootan	USA	Design and Fabrication of Closed Loop Systems (CLS) for the Fast Flux Test Facility (FFTF)
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ABBREVIATIONS

AHX	air heat exchanger
ASTRID	advanced sodium technological reactor for industrial demonstration
CBR	core breeding ratio
CDF	cumulative damage fraction
CEA	French Alternative Energies and Atomic Energy Commission
CEFR	china experimental fast reactor
CNFC	closed nuclear fuel cycle
CORAL	compact reprocessing of advanced fuels in lead cell
CPNER	numerical and experimental research
CR	control rod
CSRs	control and safety rods
CTMS	cladding tightness monitoring systems
DAE	Department of Atomic Energy
DFRP	demonstration fast reactor fuel reprocessing plant
DHRS	decay heat removal system
DSR	diverse safety rod
ECCS	emergency core cooling systems
EURATOM	European Atomic Energy Community
FAST	floating absorber for safety at transient
FBR	fast breeder reactors
FBTR	fast breeder test reactor
FFF	fuel fabrication facility
FFTF	Fast Flux Test Facility
FHX	forced-draft sodium-to-air heat exchanger
FR	fast reactor
FRF	fuel recycling facility
FRFCF	fast reactor fuel cycle facility
GFR	gas cooled fast reactor
GIF	Generation IV International Forum
I&C	instrumentation and control
ICT	integrated cold trap
IGCAR	Indira Gandhi Centre for Atomic Research
IHTS	intermediate heat transfer system
IHX	intermediate heat exchangers
INPRO	innovative nuclear reactors and fuel cycles
ISAM	integrated safety assessment methodology
JAEA	Japan Atomic Energy Agency
KAEC	Korean Atomic Energy Commission

KAEPC	Korea Atomic Energy Promotion Council
KAERI	Korea Atomic Energy Research Institute
LCOE	levelized cost of energy
LFR	lead cooled fast reactor
LUFC	levelized unit fuel cost
LWR	light water reactor
MOX	mixed oxide
MSR	molten salt reactor
MTR	material test rig
MTR	materials test reactors
NFC	nuclear fuel cycle
NRV	non-return valves
NSSS	nuclear steam supply system
P&T	partitioning and transmutation
PBT	potential biological toxicity
PDEC	pilot demonstrator energy complex
PDHR	passive decay heat removal systems
PFBR	Prototype Fast Breeder Reactor
PGSFR	Prototype Generation IV SFR
PIE	post-irradiation examinations
PSAR	preliminary safety analysis report
PSID	preliminary safety information document
PSP	primary sodium pumps
R&D	research and development
RFEC	remote field eddy current
SAFE	static absorber feedback equipment
SC	steering committee
SDC	safety design criteria
SDG	safety design guidelines
SDS	shutdown systems
SDSAR	specific design safety analysis report
SFR	sodium cooled fast reactor
SmART	small amount of reused fuel test
SPFF	system of passive flow feedback
SSC	structure, system and component
SSKGO	system of fuel element claddings
STELLA	sodium thermal-hydraulic test program
SWRPRS	sodium-water reaction pressure relief system
TBP	tri-n-butyl phosphate

TEPCO	Tokyo Electric Power Company
TiAP	tri-isoamyl phospahte
TOP	transient over-power
UCS	upper core structure
ULOF	unprotected loss of flow
V&V	validation and verification
VHTR	very-high-temperature reactor
VVER	water-water energetic reactor
YGE	young generation event

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