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PROCEEDINGS SERIES

Fast Reactors and Related Fuel Cycles: Sustainable Clean Energy for the Future (FR22)

PROCEEDINGS OF AN INTERNATIONAL CONFERENCE
Vienna, Austria, 19–22 April 2022

FAST REACTORS AND RELATED FUEL CYCLES: SUSTAINABLE CLEAN ENERGY FOR THE FUTURE (FR22)

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PROCEEDINGS OF AN INTERNATIONAL CONFERENCE
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FOREWORD

In 2021, the world celebrated the 70th anniversary of electric power production by a fast reactor. Today, the benefits of fast reactor power generation are more relevant and imperative than ever. Fast reactors operating in a closed fuel cycle allow for the efficient use of nuclear fuel, preserving natural resources and reducing the burden of generated wastes, and for the production of reliable, low carbon power. Recently, several prominent fast reactor projects have made significant progress in safety, reactor design maturation, fuel support facilities, and test or demonstration facility development. Advances in modelling and simulation have enabled large strides in both design and licensing activities. For these reasons there has been a resurgence in interest in advanced nuclear technologies.

IAEA activities in support of fast reactors and related fuel cycle activities are guided and implemented by both 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). These groups of experts meet annually to review IAEA activities, share the status of activities from their countries and provide guidance for planning future activities to best address future needs.

An important IAEA activity in support of fast reactor technology development is the recurring international conference series on the topic. The first International Conference on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities, was held in Kyoto, Japan, in 2009. This was followed in 2013 by a second conference, held in Paris, spotlighting Safe Technologies and Sustainable Scenarios, and in 2017 by a third conference, held in Yekaterinburg, Russian Federation, on Next Generation Nuclear Systems for Sustainable Development.

The fourth conference — originally to be held in 2021 in Beijing but postponed owing to restrictions relating to the COVID-19 pandemic — was held in 2022 in Vienna. Its theme of Sustainable Clean Energy for the Future highlighted the suitability of fast reactors for achieving global net zero goals aimed at decarbonizing electricity production. The conference provided a valuable forum for information sharing and capacity building.

These Proceedings provide a summary of the different technical, plenary and panel sessions; the full papers and presentations submitted to the conference are available on the INDICO platform on the IAEA Fast Reactor 2022 conference web site.

The IAEA is grateful to the International Advisory Committee, the International Scientific Programme Committee and the Secretariat of the Conference for their commitment and support.

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1. SUMMARY

1.1. INTRODUCTION

The International Conference on Fast Reactors and Related Fuel Cycles: Sustainable Clean Energy for the Future (FR22) organized by the International Atomic Energy Agency (IAEA) once again brought together the fast reactor and related fuel cycle communities. The conference was held in Vienna, Austria from 19-22 April 2022. Originally set to continue the four-year cycle, the conference was planned to be held in Beijing, China in May 2021.

Due to the COVID-19 pandemic and the prioritization of the health and safety of both the conference participants and the host country, it was jointly decided to postpone the conference to 2022 and for the IAEA to host in Vienna. Furthermore, the conference format was adapted to allow virtual participation, to ensure accessibility for all interested participants. This format change was done by making all sessions available virtually for registered participants and observers, exhibiting virtual narrated poster presentations, and by adapting the conference programme to the new circumstances from the originally drafted programme.

FR22 follows the successful first International Conference on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09) held in Kyoto, hosted by the Government of Japan; the second conference, spotlighting Safe Technologies and Sustainable Scenarios (FR13), held in Paris, hosted by the Government of France and the last conference, FR17, hosted by the Government of the Russian Federation in Yekaterinburg. FR17, themed as Next Generation Nuclear Systems for Sustainable Development.

Currently, there are several fast reactors already operating, with more under construction, and new innovative concepts under development, highlighting the importance of fast reactors and related fuel cycles in ensuring the long-term sustainability of nuclear power. Fast reactors can extract almost all energy contained in uranium, with fast breeder reactors allowing for even better resource management and contributing to a significant reduction in the burden of the radioactive waste.

The conference was structured to cover major technical aspects and topic involved in fast reactors and related fuel cycle technology such technology advances, related safety, economics, proliferation resistance and sustainability.

1.2. SUMMARY OF THE CONFERENCE

The purpose of the conference was to provide a valuable opportunity to exchange and distil information on fast reactors and related fuel cycles. The conference served to highlight the status of national and international programmes, as well as identify the challenges facing the technology. The conference was organized into nine thematic tracks as follows:

- Track 1. Innovative Fast Reactor Designs
  - Basic reactor and core designs
  - Sodium cooled fast reactor (SFR) design features
  - Lead cooled fast reactor (LFR) design features
  - Accelerator-driven systems (ADS) design features
  - Fast neutron microreactors
  - Other fast reactor designs
  - Fast reactors for cogeneration and integrated energy systems
  - System and component design and qualification

- Track 2. Fast Reactor Safety
    - Safety by design approaches, passive safety systems
    - Harmonization of safety requirements
    - Core disruptive accidents
    - Probabilistic safety assessment
- Track 3. Fuels, Fuel cycles and Waste management
    - Development of innovative fuels: Design, manufacturing, thermo-physical properties, irradiation experiments and post irradiation examinations of advanced fuels
    - Fuel performance and qualification
    - Closing the fuel cycle (actinides multirecycling, partitioning and transmutation (P&T), etc.)
    - Reprocessing and partitioning
    - Waste minimization and approaches to manage generated waste
    - Safety of processes and facilities
- Track 4. Fast Reactor Materials (coolants, structures) and components
    - Advances in coolant technology
    - Innovative structural materials
    - Component technology
- Track 5. Test Facilities and Experiments
    - Experimental facilities under design, construction or operation
    - Experimental tests and results
    - Remaining gaps and needs
- Track 6. Modelling, Simulations and Digitalization
    - Neutronics simulations
    - Advances in fast reactor thermal hydraulics
    - Coupled and multiphysics simulations
    - Verification and validation
    - Uncertainty and sensitivity analysis
    - Digitalization of reactor models
- Track 7. Sustainability: Economics, environment and proliferation
    - Sustainability of innovative fast neutron systems
    - Public support
    - Integration with other energy sources (hybrid energy systems)
    - Non-proliferation issues and proliferation-resistant-by-design concepts
- Track 8. Commissioning, Operation and Decommissioning
    - Commissioning and operation of SFRs
    - In-service inspection and repair, instrumentation
    - Decommissioning (including management of waste arising from decommissioning)
- Track 9. Education, professional development and knowledge management
    - Education and training: national and international programmes
    - Knowledge preservation and transfer
    - Projects on international collaboration, mobility and capacity building
    - Basic principles simulators for educational purposes

Major advances in several key areas of technological development were present at 26 technical sessions by approximately 200 oral presentations. Additionally, 65 posters were presented virtually and complemented the worldwide technical and scientific developments. All these technical papers were peer-reviewed by the International Scientific Programme Committee members and revised accordingly by the authors.

The conference also included three panel events devoted to the national and international fast reactor programmes; designs, applications, and fuel cycles of innovative fast reactors; and empowering the next generation's professionals: towards gender balance and cross-cutting disciplines. A young generation event (YGE) with presentations from the young professionals who won the YGE essay contest was organized as part of the plenary session.

1.3. OBJECTIVES AND STRUCTURE OF THE PROCEEDINGS

These Proceedings are intended to summarize the presentations and discussions that took place during the FR22 conference and serve as a source of information for newcomers and specialists involved in fast reactor technology and related fuel cycles developments. The structure is set to mirror the structure of the conference, as each section represents a technical session and includes summary, highlights, and challenges of the presented papers. In addition, opening session, summary of the plenary sessions highlighting the national and international programmes developments, summary of the thematic panel sessions, young generation event and closing session are also included. Because of the hybrid nature of the conference, there was no dedicated poster session. Instead, the poster presentations were available throughout the conference through the IAEA Conference application and displayed as electronic e-posters during the conference on large screens outside of conference rooms. Thus, the poster presentations are reported in these proceedings per thematic track. Individual papers, recordings, and presentations are available through the IAEA Conference application. The objectives of the conference and proceedings were to:

- Provide a platform to foster information exchange between Member States with advanced reactor programmes;
- Highlight the agency's activities in support of fast reactors and related fuel cycles;
- Discuss the status and specific challenges facing technical areas such as materials, modelling and simulation, fuels and innovative systems, and others;
- Identify gaps in technical capability and capacity development to inform future agency activities.

2. OPENING SESSION

2.1. OPENING ADDRESS, IAEA DIRECTOR GENERAL, RAFAEL M. GROSSI

Good morning, everybody. I am delighted to be here with you for this very important conference. Hardly a day passes without us listening more and more to how important nuclear energy, nuclear technology evidently is in answering many of the challenges we are having. In this community many knew it for a long time, but still there were doubts and still there were concerns. Now we seem to be having a very interesting turn of events, which was imposed by the very evident problem of global warming. The challenges that bring climate change. Many around the world are looking back at our technology, to this industry and scientific endeavour where all of you and all participants of this conference have been making so many contributions. So, it is important that we continue to show the progress, the forward-looking nature of nuclear technology, addressing like we are doing today this important area of work.

This is the fourth edition of the International Conference on Fast Reactors and Related Fuel Cycles and follows on the successes that we had in Kyoto, in Paris and also in Yekaterinburg and also, we are looking forward to next one, that will take place in China. Finding a more sustainable way to fuel the global growth and prosperity is of course one of the biggest challenges that we are facing, and I was referring to that just a second ago. To make this transition without major interruptions or price spikes like the one we are all experiencing right now. We will need all low carbon energy sources, including nuclear. In fact, we are going to be needing more, indeed a lot more nuclear if we are going to power economic growth and at the same time achieve the net zero greenhouse gas emissions by mid-century. I was few months ago in Glasgow for COP26 and I intend to be present at COP27 in Egypt, Sharm el-Sheikh, to discuss again, to put again nuclear at the table where it rightly belongs. The case for nuclear power has been strengthened and we have seen the last report released by the Intergovernmental Panel on Climate Change earlier this month, explicitly mentioning nuclear, something that had not happened before, so it is perhaps slow coming realization of something we, as I was saying, all knew. Things will not happen until we harness the necessary technological, scientific, financial, political, societal resources that are needed for this to happen. Simple as that. Conferences like these are so important, because of the need to show that the technological base is there to continue moving forward in an efficient way.

Fast reactors, in fact, are nothing new. You know it very well. There were among the first technologies deployed during the early days of nuclear power, but new concepts, technologies and advances in material research combined with the long-term vision of nuclear power as part of the sustainable energy are now reviving the fast reactor option. People are looking at it with increased interest.

We have seen a lot of exciting developments on fast reactor coolants, which include sodium, lead, lead bismuth, molten salt and gas.

As the world faces these new situations, as I was saying, it is time to focus again on the enduring appeal of fast reactors and related fuel cycles. While only a fraction of natural uranium present in the fuel is used in existing thermal reactors, as you know, fast reactors can use almost all uranium in the fuel. They extract up to 70 times more energy, thereby preserving natural resources and reducing the environmental impact of mining. Besides being low carbon, like all nuclear reactors are, fast reactors systems tick key boxes when it comes to sustainability. They shrink the environmental footprint of the waste, while extracting significantly more energy from the fuel. This is driving the development of fast reactors. They can be a bridge to even safer and more efficient nuclear power providing sustainable clean energy for generations.

Fast reactors are truly a global phenomenon with countries including Russia, China, France, the United States, the Republic of Korea and Japan active in the field. The most mature fast reactor technology, the sodium cooled fast reactors, has more than 450 reactor years of experience acquired though the design, construction and operation of the experimental prototypes, demonstration, and also commercial units in a number of countries, including China, France, Germany, India, Japan, the Russian Federation, the United Kingdom, Kazakhstan and the United States of America.

As you know, the agency supports the development and deployment of fast reactor systems through sharing information and experience including coordinated research projects, technical publications, technical meetings and international conferences like this one.

Our Technical Working Group dedicated to fast reactors is celebrating 55 years of work, this year. I wish to congratulate the successive members of the group, which have done so much in this area.

The IAEA's International Project on Innovative Reactors and Fuel Cycles, or INPRO, also plays a role in the development and deployment of these types of innovative reactors and related fuel cycles through promoting institutional approaches and international collaborations to enhance nuclear energy sustainability.

Of course, everybody talks about small modular reactor these days and many of those and micro reactors under development are fast reactor designs as well. It is good to be reminded of this.

I would like to refer to something that some of you may be aware of. It is my newly launched Nuclear Harmonization and Standardization Initiative for the safe and secure deployment of SMRs (NHSI), where many of most important companies in the world and all nuclear regulators in the world are involved bringing together stakeholders from across the world to speed up the pace at which this vital technology can be deployed with all the requisite safety and security measures, and this goes back to my initial comment. We may know that we have a good technological solution, but we need to overcome some problems, some difficulties and make our work more efficient. This is why I launched NHSI, which is looking at the standardization side of the thing, from the industry, and we are working with the industry, it is not us telling them what they need to do because they know better. But they are working with us and also, the indispensable collaboration of our regulators. They need to be there because licensing in a world of small modular reactor is going to be different. This is what NHSI is all about. International cooperation is needed to catalyse innovation and support Member States in addressing outstanding challenges to the wide deployment of fast reactor systems and technologies.

This conference I am sure will bring us to the best state of the art and to the most interesting aspects of work, which has been carried out in all of our nations. In keeping with the previous events, this week's conference will feature a Young Generation Event and winners of the Essay Competition will have the opportunity to present their innovations. I would like to emphasize that this is not a side show, it is a very essential part of the conference.

FR22 stands not only for the development of the next generation fast reactor systems, but also for the development of next generation of scientists and engineers and for promoting the participation of female professionals as the IAEA remains committed to achieving gender balance in all areas of the nuclear energy field. You may have heard of my Marie Skłodowska Curie Fellowship, of which already more than 200 women in less than 2 years have been benefiting and we are aiming for more. This is a concrete accomplishment apart from the exhortations from a podium, which the IAEA is really bringing women, opening doors for them, bringing opportunities for them to join in the workforce and in the research force.

This conference, which is about a technology which is not new, but is every bit as promising as it first was when launched few years ago, has together with nuclear an important opportunity to be what at the end of the day this is all about, improving people's lives. Thank you very much.

2.2. CONFERENCE CHAIR ADDRESS, ARUN KUMAR BHADURI

A very good morning to everybody and thank you Director General Grossi for making my opening address a little simpler. Nevertheless, on behalf of the fast reactor community, I would like to re-iterate many of the things that you already mentioned, maybe with some facts and some figures.

Fast reactors are essential for energy security and for the environment. Although the other carbon footprint sensitive celebrities got more subsidies than we have got for the nuclear, it is again public perception which the Director General was referring to. We have to take the societal part into account. We believe that nuclear energy should be embraced, from both the energy security as well as the environmental perspective because it is one of the least expensive forms of power, it releases no polluting gases, we all know that and we need to reiterate that, and use of land is very limited. Typically, a double reactor plant may take few hectares, which can power about 2 million homes, but the same production if you have to use the more commonly known wind or solar will take tens of thousands of hectares.

There have been concerns regarding nuclear waste but with modern techniques the spent nuclear fuel is safely removed with the guidance from the Agency and reprocessed to yield new reactor fuel. This drastically reduces the amount of nuclear waste.

We should have not nuclear alone, but a mix of energy basket. The safety levels of nuclear energy productions surpass other energy sources many folds. It is not accepted societally but it is a fact if you compare them to the thermal coal fired plants for example. According to some of the top conservation scientists, who are independent, they say this is the greenest form of energy and one of the most environmentally safe sources of energy and should be accepted to avoid future energy crisis.

A careful objective analysis has shown that nuclear power should be taken from the point of view of cost, point of safety, emission reduction, land use and pollution, all combined together. If you looked at the full life cycle emissions per GW of power, you would find that solar photovoltaic are almost double that of nuclear, hydro, geothermal and wind. We have not been able to highlight this at various forums. If we use photovoltaic, we only see what it emits during the production of electricity, we do not see what it takes to make a solar cell or to dispose of the photovoltaic cell. This is one area which we need to demonstrate and to reflect to all communities that it is indeed the way that the Director General presented at COP26.

The role of fast reactors comes in various forms. The fast reactors should be, in principle, able to extract almost all energy contained in uranium. The light water reactors (LWR) use a percent of the energy that is available in uranium, 99% of uranium is not used, it is a resource not being utilized. The efficiency of fast reactors is also much better, around 40%, because they are high temperature systems. Whereas the thermal reactors operate at 300-350°C, the fast reactors operate at 500-600°C and might go even higher. Therefore, in one of the Standing Advisory Group for Nuclear Energy (SAGNE) meetings it was said that fast breeder reactors are effectively a renewable form of energy. We use the fuel, reprocess it and that is why the fast reactors and reprocessing technologies are taken together. They have to be taken in a holistic manner.

Fast breeder reactors, though most people use ‘fast reactors’, we in many countries look at fast breeder reactors because it is the only way of better resource management, will take for few hundreds of years and provide the long-term energy supply. Importantly, the waste management which has been a concern, might be solved by incineration of the minor actinides. About 4-5% of minor actinides can be added to the nuclear fuel without affecting the efficiency of the repowered reactor. And therefore, it will reduce the long-term storage requirements. In the fast reactor programme, it is essential to have a closed fuel cycle. To use the initial fuel, replenish the used fuel with fresh uranium and continue to reuse it. This way 70-80% of the energy available in the uranium that is mined with so much effort will be utilized.

If we look at the growth of nuclear power beyond the light water reactors and heavy water reactors, high growth rate is possible if we use breeders with high breeding ratio. If we look at the burn up of the fuel, a natural uranium pressurized heavy water cooled reactor or the enriched uranium thermal reactor have a burn up of typically 30 to 40 GWd/t, whereas for the fast reactors, we go up to 100-200 GWd/t of fuel, which means a residence time of 2-3 years in the power reactor. Still better, the breeding ratio and doubling time with the oxide fuel is somewhere around 28-35 years. If we switch over to the metallic fuels, the breeding time comes drastically down to about 10 years, which is phenomenal. Thus, lots of work has to be done in transitioning from the type of fuels currently in use such as the oxide, the carbide or the nitrides to metallic fuels. This is essentially required to grow the clean energy capacity necessities going for nuclear in a larger way and flagging it off as a renewable form of energy, like solar or wind.

Effective incineration is done with fast reactors in high energy spectrum. If we are looking at recycling of waste, if we do not use the minor actinides or the long-lived isotopes, once we put the waste in deep repositories it will take about 100 000 years for the radioactivity to come down to the background of the mining area. If we now use the fuel with 4-5% minor actinides, the waste activity in repositories will take about 200 years to come down to the background levels of the mine, which is really a selling point when we look at how to store the nuclear waste.

The last thing which the Director General mentioned, and we recommended during a recent SAGNE meeting: the IAEA has made great appreciable efforts to advocate for the indispensable role which nuclear power energy plays and will play in fighting the climate change and I believe this FR22 conference is one of those efforts. With this I also join all of you in wishing for a very successful FR22.

3. SUMMARY OF THE PLENARY SESSIONS

3.1. CHINA

H. Yang, Vice President of the **China Institute of Atomic Energy (CIAE)**, explained that nuclear energy is an important approach for China to carry out the international commitment for reducing carbon emissions. China's nuclear power generation capacity will increase from current 4.8% to 10% and reach 150 GW around 2035. The Strategy is to develop fuel cycle along with the nuclear power stations. The R&D of Small modular fast reactors in under development in parallel. China aims to develop commercial reactor called CFR1000 by 2030. Demonstration reactor, called CFR600, is a typical pool type SFR with 2 symmetry circuits and 8 modulars SGs in every circuit and its power is 1500 MW(th) and 600 MW(e). The computer codes needed for the SFR design and safety analysis have been already developed (especially for CFR600). He presented the basic concepts of Integrated FR Nuclear Energy System (IFRES): composed of several fast reactors and one fuel regeneration facility at the same site to integrate fuel cycle processes.

3.2. FRANCE

F. Serre, Deputy of the **Nuclear Energy Division at the French Alternative Energies and Atomic Energy Commission (CEA)**, introduced the "Multiannual Energy Plan". France remains committed to the closed fuel cycle policy until 2040. The French president announced the Investment Plan for Green Energy. The plan consists of three sectors; to develop SMR technology with better waste management, to mass-produce green hydrogen and to reduce greenhouse gas emission. The French Projects focus on SMR as well as Advanced Modular Reactors (AMR) such as SFR. The objective of AMR-SFR programme is to assess the opportunities in terms of safety, economy and fuel cycle. France also has interest in fast Molten Salt Reactors (MSR) because of its potential assets, such as in the nuclear fuel cycle with the transmutation of MA and multi-recycling of plutonium, intrinsic safety and flexibility. However, MSRs also face feasibility issues such as salt chemistry, materials and safety in operation. France started the R&D programme in 2020 that aims at assessing the feasibility of fast MSRs and confirm their potential assets. This is complementary with the main, ongoing R&D on Gen IV SFRs.

3.3. INDIA

D. Venkatraman, Director of the **Indira Gandhi Centre for Atomic Research (IGCAR)**, summarized the characteristics of Fast Reactors and explained the R&D status of the Fast Breeder Test Reactor (FBTR). The three-phase Indian Fast Reactor Programme consists of Phase I: FBTR, which has been successfully accomplished in 2022 when the FBTR attained rated capacity of 40 MW(th); Phase II focused on PFBR and on techno-economic demonstration, which is in progress and expected to be finalized by 2030 and Phase III commercialization with FBR-MOX reactor by 2047. Various studies were carried out in India to extend the life of FBTR and it is expected to operate until 2034 or 2035. India continues to develop reprocessing technology using pyrochemical technique based on molten salt electrorefining. Reprocessed Pu was refabricated into fuel and was generating power in FBTR, thus demonstrating the closure of FBR fuel cycle. Further works on fuel cladding behaviour were reported.

3.4. JAPAN

K. Hideki, **Japan Atomic Energy Agency (JAEA)** introduced "The 6th Strategy Energy Plan of Japan" that was approved by the Cabinet in October 2021: it states the utilization possibility of nuclear energy for carbon neutrality. Nuclear Energy X Innovation Promotion (NEXIP) is a new initiative intended to accelerate the development of nuclear innovative technologies in Japan. Through NEXIP and other programmes, FR technologies and international cooperation projects are supported by Government. As a result of reactor design R&D, Japan proposes a pool-type reactor with a 3D seismic isolation and develops ARKADIA as a digital triplet tool for reactor design. JAEA also contributes to new codes and standards for flexible design in ASME. Mr. Hideki reported on Japanese experimental facilities such as Joyo, an experimental fast reactor in which neutron irradiation experiments are conducted. Japan will promote the reduction of environmental burden in terms of improvement of the flexibility in Pu use, and verification of Minor Actinide partitioning and transmutation technologies.

3.5. REPUBLIC OF KOREA

C. Lim, **Korea Atomic Energy Research Institute (KAERI)**, informed that the Korean Government has an interest in Sodium cooled Fast Reactors (SFR) coupled with pyro-processing as part of pending issues with long-term Spent Nuclear Fuel (SNF) management. The Korean SFR programme has been steadily moving toward technology demonstration of TRU transmutation as well as SFR safety, but specific plans for deployment need to have further decision on the pyro-processing technology development. Prototype Generation IV Sodium cooled Fast Reactor (PGSFR) Project has been developed by KAERI with the grant from Korean Government since 2012 and is currently waiting for further Government decision to deploy domestic TRU transmutation demonstrator as one of the key SNF management options. SFR-based advanced SMR is also being developed with spin-off technologies and feedback from the PGSFR programme. KAERI is expecting valuable technology impacts through an opportunity of international cooperation in the field of SFR design and correlated R&Ds.

3.6. RUSSIAN FEDERATION

V. Pershukov, Special Representative of **ROSATOM** for International and Scientific Projects, delivered that New Technology Platform (NTP) of Russia consists of two major elements: Fast Reactor (FR) and Closed Nuclear Fuel Cycle technologies (CNFC). Russia has vast fast reactor operation experience with plans to expand its FR fleet in the future. Two FRs with a sodium coolant are currently in operation (BN-600 and BN-800). Russia is developing breakthrough reactor and closed nuclear fuel cycle technologies addressing key issues such as inherent safety, unlimited fuel supply, waste minimization and economic competitiveness. The purpose of the Pilot Demonstration Energy Complex (PDEC) is to demonstrate the feasibility of an inherently safe nuclear reactor with a closed nuclear fuel cycle. PDEC will feature state-of-the-art on-site fuel fabrication and reprocessing capabilities. Current progress on closed nuclear fuel cycle technology development is comprised of three key elements – manufacture, experiment and demonstration of mixed nitride fuel from Spent Nuclear Fuel (SNF). Considering FR nuclear fuel cycle, minimizing the wastes from SNF is a priority for closed NFC energy systems. Russian Federation is committed to raising awareness on the significance of FRs and closed NFC technologies.

3.7. UNITED STATES OF AMERICA

A. Caponiti, Deputy Assistant Secretary for Reactor Fleet and Advanced Reactor Deployment in the Office of Nuclear Energy in **Department of Energy (DOE)** highlighted that advanced reactors are crucial for achieving national and global carbon reduction goals and will grow through interaction with other energy sources for a net-zero future. The USA continues to support programmes in advanced nuclear development and is committed to 100% clean energy in the grid by 2035. The US

Fast Reactor Research programme aims to support technical and experimental R&D. Gateway for Accelerated Innovation in Nuclear (GAIN) programme is made to connect the public with nuclear expertise and create opportunities to accelerate deployment. The National Reactor Innovation Centre (NRIC) attempts to demonstrate the ability of the advanced nuclear power plant and try to improve the technology. Private companies in the USA are working toward diverse, new generation nuclear power plants for a carbon-free future: SMR of NuScale, NATRIUM Sodium cooled Fast Reactor of Terrapower and High Temperature Gas cooled Reactor of X-energy. The USA will continue to perform fundamental R&D on advanced reactor and fuel cycle technologies to improve nuclear energy safety and performance.

3.8. EUROPEAN COMMISSION (EC)

M. Betti, Director, JRC Directorate for **Nuclear Safety and Security of the European Commission (EC)**, explained that the JRC assists in advanced reactor research activities. To enhance the design and system integration, many international corporations in Europe are researching new generation nuclear power plants, such as Gen IV, and the JRC is supporting the growth of these corporations. In particular, the JRC is very interested in research activities on the structure of materials used in advanced reactors to strengthen durability and improve safety, as well as research on economic efficiency and fuel cycle safety to improve fuel quality. JRC aims to continue to make steady investments in research, training and education with related organizations and companies. All the nuclear safety research activities of JRC are embedded in partnerships and collaborations with other Directorates General of the EC, Member States organizations, networks, technology platforms, international organizations and initiatives.

3.9. GENERATION IV INTERNATIONAL FORUM (GIF)

R. Hill, Technical Director of **Generation IV International Forum (GIF)**, delivered a presentation on the GIF Goals and the recent status of developments according to the characteristics of each Generation-IV Reactor System. As GEN IV transitions from conventional reactors, the goals of GIF are to enhance the following key issues: Sustainability, Safety & Reliability, Economics, Proliferation Resistance and Physical protection. Among the six GEN IVs, Sodium cooled Fast Reactor (SFR) is still active GIF System and demonstration SFR and active R&D programmes in diverse Member countries attempt to enhance safety and improve the economics. Molten Salt Reactor (MSR) has a large interest, with more than 40 concepts of a large variety being developed worldwide. System Design Criteria (SCE) and Design Guidelines (SDG) are also being developed so that countries can use the type of reactor that best suits their needs. Currently the new initiatives to address the key GEN IV deployment issues are non-electric application of nuclear heat, advanced manufacturing and materials engineering, and continued education and training opportunities, such as GIF Webinar series.

3.10. OECD/NUCLEAR ENERGY AGENCY (NEA)

T. Ivanova, Head of the Division of Nuclear Science, **Nuclear Energy Agency of Organization for Economic Cooperation and Development (OECD/NEA)**, presented the overview of NEA Fast Reactors activities. These activities include disseminating scientific knowledge and support research and development by providing, world reference collections of integral experiments and databases, software tools and calculation parameters to support validation, and state-of-the-art reports and reports on benchmark studies. Overview of further activities was presented related to reactor physics experiment, experimental infrastructure, uncertainty analysis for Sodium cooled Fast Reactor (SFR) modelling, thermal hydraulic benchmarks, fuel cycle physics and chemistry, fuel performance and safety of advanced reactor, as a form of sharing research status and encouraging collaborative research. Nuclear Education, Skills and Technologies (NEST) Framework, launched in 2019, a multinational framework to develop skills and nurture the next generation of nuclear subject matter experts through transfer of practical experience and knowledge as well as Global Forum on Nuclear Education, Science, Technology and Policy, launched in 2021, providing academic institutions around the world with a framework for interaction, co-operation and collective action were also presented as a part of the NEA training and education activities.

3.11. INTERNATIONAL ATOMIC ENERGY AGENCY (IAEA)

A. des Cloizeaux, the Director of the **Division of Nuclear Power** in the **Department of Nuclear Energy of the IAEA**, delivered that long-term development of sustainable nuclear power will require fast reactors. Ms. Cloizeaux presented the role of the Technical Working Groups (TWGs) on Fast Reactors and Related Fuels and Fuel Cycles. TWGs provide advice to DDG-NE on specific topics of relevance to the IAEA's programmatic activities and share information and knowledge on national and international programmes. The IAEA organizes coordinated research projects on fast reactors technology with the aim to bring together research institutions from its developing and developed Member States to collaborate on research projects of common interest. The overview of recently completed, ongoing, and future projects was given. Furthermore, several workshops, training course series and simulator tools as part of the IAEA's Training, Education, and Engagement activities, and well as other events such as technical meetings were presented. The International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) provides analysis and assessment in support of long-term planning for sustainable development of nuclear energy, consists of the process of establishing a strategy of global scenarios, innovation, sustainability assessment and strategies and outreach.

TABLE 1. PRESENTATIONS FROM KEYNOTE ADDRESS

Chairs: Aline des Cloizeaux (IAEA) and Vladimir Kriventsev (IAEA)

ID	Presenter	Designating Member State/Organization	Keynote Address
K-1	H. Yang	China	Current status of China fast reactor development
K-2	F. Serre	France	Fast neutrons reactor development in France
K-3	B. Venkataraman	India	Fast reactor program in India
K-4	H. Kamide	Japan	Current status of fast reactor developments in Japan
K-5	L. Chae Young	Republic of Korea	Status of fast reactor Development program in Korea
K-6	V. Pershukov	Russian Federation	A new technological platform for global development of next generation nuclear power
K-7	A. Caponiti	USA	Overview of U.S. advanced reactor programs
K-8	M. Betti	European Commission (EC)	Development of safe and sustainable fast reactor systems: EU context and JRC contributions
K-9	R. Hill	GIF	Generation-IV International Forum
K-10	T. Ivanova	OECD-NEA	Overview of NEA fast reactors activities
K-11	A. des Cloizeaux	IAEA	IAEA activities on fast reactors technologies and related fuel cycles

4. 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.

Special Session dedicated to IAEA CRPs
Mr. Kriventsev opened the session. The Special Session comprised of seven presentations: one from the IAEA, one from the Republic of Korea, one from Mexico, three from the USA and one from China.

N. Morelová (IAEA) delivered the first presentation introducing the on-going coordinated research projects on the Fast Reactor Technology development; neutronic benchmark of (China Experimental Fast Reactor (CEFR) Start-up Tests and Benchmark Analysis of Fast Flux Test Facility (FFTF) Loss of Flow without Scram Tests. Ms. Morelová described division of the work packages of CEFR benchmark, summarised participation of the project (30 participating organizations from 18 countries) and delivered a timeline, with expected publication of the CEFR related TECDOC end of 2022. A separate TECDOC covering the work package related to Analysis of Uncertainties is expected in 2023. Similarly, the FFTF benchmark overview was given (24 participating organizations from 13 countries) and timeline presented, with the FFTF TECDOC finalization expected in 2023.

X. Huo (China) presented CEFR physical start-up tests: the core specifications and experiments. Detailed descriptions of the start-up tests, and benchmark specifications were provided for the 7 work packages: fuel load and criticality test; control rod worth measurement; temperature reactivity coefficient; sodium void reactivity; core subassembly exchange reactivity effect; reaction rate distribution; reactivity coefficients and kinetic parameter. Mr. Huo proposed a list of possible extension work, including dynamic simulation (rod drop transient), burn-up calculations and detector response factor (influence to the detector), spectrum of typical positions, comparison of group XS generated, comparison of calculated beta effective and nuclear data adjustment based on the experiment.

R. Lopez on behalf of the author Mr. **Armando Gomez Torres (Mexico)** delivered a presentation on the verification and validation of neutronic codes using the start-up fuel load and criticality tests performed in the China Experimental Fast Reactor. The CEFR core was described, with net criticality being the output of interest. Participating organizations using deterministic and stochastic neutronic codes were introduced. The blind results and refined phase results of K effective, work package 1 of the CRP were shown. In the blind phase, the deviations in pcm from average values were up to 2000 pcm between participants using deterministic codes, while in the case of stochastic codes, the deviation in pcm from average values ranked in ±400 pcm (800 pcm in the most extreme cases). These values improved in the refined phase, with deviations only ±320 pcm from experimental data with deterministic codes due to the adjustments in XS's generation and developments in the numerical solvers and ±167 pcm with stochastic codes due to model adjustments.

J. Choe (Republic of Korea) presented the results of work packages 3, 4 and 5 of the Neutronics Benchmark of CEFR Start-Up Tests: Temperature Reactivity Coefficient, Sodium Void Worth, and Swap Reactivity. The calculation approach was explained. The reactivity coefficients were mostly in good agreement with the experimental data, but the absolute mean value of the calculated results was lower than in the experiment for all three cases.
T. K. Kim (USA) presented the results of the work packages 6 and 7 of the Neutronics Benchmark of CEFR Start-Up Tests: Reaction Rates and Reactivity Coefficients. For reaction rate measurements,

both deterministic and stochastic results give good agreement in the fissile zone, but disagreement between participants and deviation from measured values increase in non-fissile zones were observed. Extra attention is needed in simulation of reactions with low cross section (^{27}Al), reactions far from the fissile zone, if the reaction rate is a threshold one (^{237}Np or ^{238}U) and reactions with strong thermal resonance (^{197}Au). For integral reactivity coefficients measurements, except for several outlier results, predicted integral reactivity coefficients by participants were generally comparable regardless of deterministic and stochastic calculations.

A. Moisseytsev (USA) presented blind phase results for transient simulations of the FFTF Loss of Flow Without Scram test #13. The FFTF facility and passive safety testing programme were outlined. The benchmark parameters to calculate and the respective results were shown: the total power, decay heat, net reactivity, peak fuel temperature, primary loop mass flowrate, coolant outlet temperatures and the primary loop hot leg temperature. Blind phase results provided confidence that many participants captured the transient progression of the test well including key information for improving transient predictions. The improvements of the models are expected during the open phase focused on reducing discrepancies between measured and predicted results.

N. Stauff (USA) presented blind phase results of the FFTF Neutronic Benchmark. The neutronic benchmark, participants and diversity of codes, methods and nuclear data libraries were shown. Relatively good agreement was reported on k-eff (close to 1.00 – good agreement with a critical core), kinetics parameters, Doppler coefficients, and rod worths between participants. Outliers may be due to different methods and nuclear data libraries or modelling inconsistencies. Larger discrepancy was reported on other reactivity coefficients (axial/radial, GEMs). Results on structure and sodium density coefficients were reported as widely spread-out. The GEM coefficient and its agreement between the participants was reported as crucial for the follow-up safety analyses. The open phase of the CRP will allow for improvements of the models.

TABLE 2. PRESENTATIONS FROM SPECIAL SESSION DEDICATED TO IAEA CRPs

Chairs: Vladimir Kriventsev (IAEA) and Nikoleta Morelová (IAEA)			
ID	Presenter	Designating Member State/Organization	Title of the Paper
285	N. Morelová	IAEA	Neutronics Benchmark of CEFR Start-Up Tests: An IAEA coordinated research project
104	X. Huo	China	CEFR physical start-up tests: the core specifications and experiments
163	R. Lopez	Mexico	Verification and validation of neutronic codes using the start-up fuel load and criticality tests performed in the China Experimental Fast Reactor
281	J. Choe	Republic of Korea	Neutronics Benchmark of CEFR Start-Up Tests: Temperature Reactivity Coefficient, Sodium Void Worth, and Swap Reactivity
233	T.K. Kim	USA	Neutronics Benchmark of CEFR Start-Up Tests: Reaction Rates and Reactivity Coefficients
534	A. Moisseytsev	USA	Blind phase results for transient simulations of the FFTF Loss of Flow Without Scram test #13
536	N. Stauff	USA	Blind-Phase Results of the FFTF Neutronic Benchmark

4.1. TRACK 1 – INNOVATION FAST REACTOR DESIGNS

4.1.1. Session 1.1. Overviews and Fundamentals of Fast Reactors

Eight presentations were given in succession in Session 1.1 followed by an open discussion with all participants.

S. Shepelev (Russia) showed the evolution from experimental small-scale reactors BR-5 and BR-10 available already by the end of the 1950s, up to the commercial large-scale reactor BN-1200, which is foreseen to be operational in 2031.

S. Raghupathy (India) highlighted the progress of future FBRs in India where a large experience has been gained (the 40 MW(th) FBTR reactor being operational since 1985) and where the 500 MW(e) prototype FBR is now in advanced stage of commissioning; progress both in design and R&D activities have been presented.

R. Hill (USA) presented the Advanced Reactor Demonstration programme . The 4 out of 10 U. S. Advanced Reactor Demonstration programme Awards are fast reactors, including Natrium for near-term demonstration. Contributions to the fast reactor technology are ongoing, including contributions to develop fuel cycle, safety analysis, material performance and regulatory framework in cooperation with diverse domestic and foreign companies.

A. Kato (Japan) presented conceptual design studies for the pool of SFRs to broaden not only options for reactor types in Japan but also the range and depth of international cooperation. In parallel, related R&Ds are conducted such as the 3D seismic isolation system; performances of the proposed system will be evaluated in 2023.

A. Alemberti (Italy) presented the lead fast reactor activities in GIF. The strong and friendly collaboration among the steering committee partners since 2004 is considered a major factor in the effective support to the development of LFRs. Several position papers have been published, while different reference concepts (from the small-scale SSTAR to the large-scale ELFR) have been presented and compared.

M. Caramello (Italy) presented and discussed the progress of the ALFRED lead cooled fast reactor. This reactor is to be built in Romania and the design has been improved. The licensing strategy has also been defined and a staged approach has been found as the most appropriate solution to compensate the present lack of operational experience.

D. De Bruyn (Belgium) presented the recent advances in organization and in design of the MYRRHA ADS programme in Belgium where the non-profit organization (necessary to attract potential partners) is now operational. The recent progress in the primary system (resulting in a more compact reactor vessel) has also been presented.

P. Vácha (Czech Republic) presented the research projects focused on the development of Gas cooled Fast Reactors (GFR), ALLEGRO and HeFASTo and described their position in the broader framework of the ALLEGRO development programme. The recent progress in the thermal hydraulic and helium experiments has been presented as a part of national R&D project SAFE-G PROJECT.

TABLE 3. PRESENTATIONS FROM SESSION 1.1 – OVERVIEWS AND FUNDAMENTALS OF FAST REACTORS

Chairs: Didier De Bruyn and Bob Hill

ID	Presenter	Designating Member State/Organization	Title of the Paper
136	S. Shepelev	Russia	Development of BN reactor technology in Russia
478	S. Raghupathy	India	Progress in the design and R&D for future FBRs
432	R. Hill	USA	Overview of U.S. fast reactor technology R&D program
269	A. Kato	Japan	Progress in conceptual design of a pool-type sodium-cooled fast reactor in Japan
92	A. Alemberti	Italy	Status of Generation-IV lead fast reactor activities
87	M. Caramello	Italy	The status of the ALFRED project
323	D. De Bruyn	Belgium	MYRRHA, the Belgian prototype that fascinates the world
304	P. Vácha	Czech Republic	GFR research and development programme in V4 countries

4.1.2. Session 1.2. Innovative Design Advances

Eight presentations were given in succession in Session 1.2 followed by an open discussion with all participants.

P. Ferroni (USA) presented a high-level overview of the Westinghouse Lead Fast Reactor (LFR) design and of associated development activities. Particularly, the ongoing experimental programme in support to the Westinghouse LFR in the United Kingdom, with emphasis on eight test facilities currently being set up.

V.V. Lemekhov (Russian Federation) presented an overview of the BREST-OD-300 Lead Fast Reactor, including historical information on the motivations behind the pursuit of LFR technology and details on the main engineering solutions adopted in this plant (fuel, primary system configuration and associated main components such as pumps and steam generators). The presentation also provided information on key safety analysis results and experimental programmes that have been conducted to substantiate the safety case presented and later approved by the Russian regulatory authority.

G. Toshinskii (Russian Federation) presented an overview of coolant types historically considered for fast reactors, and highlighted advantages and drawbacks of each. The material presented then focused on Lead-Bismuth Eutectic (LBE), providing some more details on this coolant including considerations on Bismuth's abundance on Earth and potential impact on cost, as well as some information on operating experience including oxide cleaning and materials corrosion performance.

T.D.C. Nguyen (Republic of Korea) presented the design and key performance characteristics of a 100 MW(e) nitride-fuelled, lead-bismuth cooled fast reactor core, adopting supercritical CO_2 power conversion cycle and a >10 year fuel cycle length, known as ANTS-100e. This work focused primarily on reactor physics and thermal hydraulic characteristics, computed using the Argonne Reactor Computation (ARC) suite of codes as well as the UNIST in-house Monte Carlo code MCS.

L. Fiorito (Belgium) presented a new neutronic design of the MYRRHA lead-bismuth cooled fast reactor core, developed to accommodate MOX fuel with a lower plutonium enrichment than earlier core versions (<30%) and a new layout for material irradiation regions. The work presented key neutronic performance indicators associated with this new core configuration, as well as a scoping study on the effect that various reflector materials have on neutron economy and on irradiation damage of the core barrel.

D. Samokhin (Russian Federation) presented the conceptual design of a small (0.5 MW(th)), lead cooled fast reactor intended for special applications such as isotope production and education / training of young professionals. The design is developed leveraging Obninsk Institute of Atomic Energy's expertise in the development of low-power reactors together with know-how of the Physics and Power Engineering Institute in the field of heavy liquid metal coolant technology.

S. Poumerouly (France) presented a multi-objective optimization analysis performed on the design of a 1000 MW(e) sodium cooled fast reactor, using the SHADOC-based Design Development System (SDDS). SDDS integrates ERANOS (for 3D neutronics), GERMINAL (for fuel assembly thermo-mechanics) and MAT5DYN (for simplified thermal hydraulics), which are run on a grid of approximately one million of possible core designs, through surrogate models generated based on the kriging method. With the ultimate goal of reducing capital cost of future French commercial SFRs while satisfying high levels of safety, the analysis identified two compact, low-sodium void reactivity coefficient, core geometries for subsequent consideration.

S. Jang (Republic of Korea) presented the conceptual design of a hybrid micro modular reactor (HMMR), which is envisioned to result from the combination between the MMR design developed by KAIST with a renewable energy and energy storage system (ESS). Specifically, the work presented a reactor physics characterization of the H-MMR, which is a 18 MW(th), ultra-long cycle (up to 56 years) fast reactor provided with a novel core configuration fuelled with uranium nitride, cooled by potassium-based heat pipes and capable for autonomous load-following.

TABLE 4. PRESENTATIONS FROM SESSION 1.2 – INNOVATIVE DESIGN ADVANCES

Chairs: Paolo Ferroni and Andrei Moiseev			
ID	Presenter	Designating Member State/Organization	Title of the Paper
528	P. Ferroni	USA	The Westinghouse Lead Fast Reactor: overview and progress in development
363	V.V. Lemekhov	Russian Federation	Pilot Demonstrational Fast Reactor with Lead Coolant BREST-OD-300
387	G. Toshinskii	Russian Federation	Choice of a coolant for a modular small power reactor SVBR-100
275	T.D.C. Nguyen	Rep. of Korea	Core Design of 100MWe Advanced Nitride-fueled Simplified Liquid Metal Cooled Fast Reactor
309	L. Fiorito	Belgium	Novel neutronics design of the MYRRHA core
219	D. Samokhin	Russian Federation	Project of a multipurpose lead reactor with a hard neutron spectrum
360	S. Poumerouly	France	Proposal of a compact core design for the 1000 MWe French commercial Sodium Fast Reactor by means of the SDDS multi-objective optimization tool
271	S. Jang	Rep. of Korea	Conceptual design of ultra-long life hybrid micro modular reactor cooled by potassium heat pipe

4.1.3. Session 1.3. System Innovations

Seven presentations were given in succession in Session 1.3 followed by an open discussion with all participants. Overall, the session covered a variety of innovative system designs / design solutions which hold promise for improving safety & economics of fast reactors.

T. Beck (France) covered an overview of the sketch design studies of the fuel sub-assemblies (S/As) for a 150 MW(e) SFR, as on end 2019. The presentation indicated that skills developed on ASTRID programme since 2010 have been efficiently employed to produce in less than 2 years, an acceptable sketch design for the reactor. It also highlighted that thermomechanical behaviour of the fuel bundle has been calculated with DOMAJEUR code. The lower gas plenum of the fuel pins has been reduced based on GERMINAL fuel performance code. The upper neutron shielding is made of steel and B4C rings housed in a leak tight compartment to stay compatible with the washing process, while limiting the secondary sodium activation and the irradiation of the diversified absorber rods electromagnet.

S. Rukhlin (Russia) presented an optimization study of built-in (in-vessel) primary sodium purification system for the advanced BN reactor plant. The unique feature of the design is elimination of the in-vessel electromagnetic devices along with its associated auxiliary circuits. The presentation highlighted some alternate technical solutions in the design of the cold trap, including the use of interchangeable valves located in the top part of cold trap and eccentric headers. The effect of optimisation resulted in reduction in the replacement frequency of cold traps during service life of the reactor plant, and associated costs of replacement through better utilisation of the cold traps.

J. Mote (India) presented the discussion of design aspects of arrangement conceived for lowering / lifting of Large and Small Rotatable Plugs (LRP/SRP) to enable replacement of the rotatable plug seals along with the results of structural analysis carried out to confirm the design. The arrangement consists of circular rings, thrust bearings, washers, nuts and connecting tie rods. The maximum allowable torque and permissible rotation of the nut during each operation in the tie rod is estimated. Structural analysis of the arrangement has been performed to ensure that the stresses induced in various parts involved are within allowable limits. The maximum allowable torque and nut turn during each operation in the tie rod were presented.

P. Kumar Patel (India) presented the design of the Secondary Sodium based Decay Heat Removal System (SSDHRS) for future Fast Breeder Reactor (FBR). Process design of SSDHRS is carried out for both forced circulation and natural circulation modes of flow through both secondary sodium and air side. For forced circulation flow, the secondary sodium main pump is used for secondary sodium flow and a blower is used for air side flow. Studies are carried out for natural circulation capacity considering, Secondary Sodium Pump (SSP) and blower in OFF condition. The heat removal capacity is estimated as 8.6 MW(th) per SSDHR loop which is about 57% of capacity achieved during forced circulation flow. With two secondary sodium loops available, SSDHR system alone would also satisfy the DHR requirements as a diverse DHR system.

M. Caramello (Italy) covered the technical feasibility of integrating a thermal energy storage in the Advanced Lead cooled Fast Reactor European Demonstrator (ALFRED) and evaluating the hourly load-following performances against wind production variations, in a reference high-penetration V-RES market. The reference case for the assessment is Romanian electricity market. The study concluded that ALFRED with molten-salt integrated TES solution has promising load-following capability that are, however, best suited to a quarter of a day basis variation, rather than hourly, as originally targeted, yet the cost-effectiveness of this solution needs to be assessed against the forecasted high penetration V-RES market demand and flexibility requirement as well as considering the increase of both capital and O&M plant cost, due to the molten-salt system.

D. Gérardin (France) presented EDF's in-house multi-physics optimization tool SDDS to define a compact core design for the 1000 MW(e) French commercial SFR. The presentation focused on SDDS study performed on a 1100 MW(e) power reactor in order to evaluate the impact of an increase of 10% of the nominal power. The analysis of the results shows that the main trends (large pellet, large fertile plate height, etc.) are the same as the ones observed in the previous study. However, a degradation of the safety performances is observed and impacts both the behaviour in case of Unprotected Loss of Service Station Power and in case of Unprotected Control Rod Withdrawal.

B. Merk (United Kingdom) covered the IMAGINE technology designed to provide the optimal approach to achieve closed fuel cycle operation while avoiding the most complex and costly components of the traditional fuel cycle. The opportunities and challenges of the project were described including the progress up to date. This state is used to describe the next steps required in a stepwise approach to develop an innovative nuclear reactor system in a structured development process characterized through basic studies, advanced studies, experimental reactor, and full-scale industrial system demonstration.

TABLE 5. PRESENTATIONS FROM SESSION 1.3 – SYSTEM INNOVATIONS

Chairs: S. Raghupathy

ID	Presenter	Designating Member State/Organization	Title of the Paper

6	T. Beck	France	Sketch Design of Fuel Sub-Assemblies for a SFR-150 MWe
190	S. Rukhlin	Russian Federation	Optimization of built-in primary sodium purification system for advanced BN reactor plant
477	J. Mote	India	Design & analysis of a novel arrangement for coupling and decoupling of rotatable plugs in PFBR
502	P. K. Patel	India	Design of secondary sodium based decay heat removal system for future fast breeder reactors
343	M. Caramello	Italy	Integration of small modular lead fast reactor with energy storage for load-following operation in high V-RES penetration electricity markets
341	D. Gérardin	France	Evaluation of an increase of the power density for the French commercial Sodium Fast Reactor and optimization study at 1100 MWe with the SDDS tool
127	B. Merk	UK	iMAGINE - a Breakthrough Technology for Closing the Fuel Cycle without Reprocessing

4.1.4. Track 1: Poster Session

Eight poster presentations were accepted for Track 1. The topics included development of nuclear burning wave reactor, specific tool development in PFBR reactor, neutronic study of the nitride fuel in the China initiative Accelerator Driven System (CiADS), modelling of the flow and temperature distribution in the ALFRED hot plenum using the Computational Fluid Dynamics (CFD) code; a study of dual-purpose fast reactor with heavy liquid metal coolant and a large-scale production of plutonium-238; study of the use of lead isotope ^{208}Pb as a neutron reflector in the fast reactor and studies focused on the metallic fuel and its breeding ratio.

TABLE 6. POSTER PRESENTATIONS FROM TRACK 1

ID	Presenter	Designating Member State/Organization	Title of the Paper
16	Y. Zhang	China	The neutronic study of the nitride fuel loaded CiADS core
122	I. Drobyshev	Russian Federation	Hybrid high power fast breeder reactor with metallic fuel and additives consisting with lightweight atoms
158	E. Kulikov	Russian Federation	On substantial slowing down of the kinetics of fast transient processes in fast reactor
159	E. Kulikov	Russian Federation	Investigation of characteristics of fast power reactor with an additional function of large-scale production of plutonium-238
365	S. Fomin	Ukraine	Power control of the fast nuclear-burning-wave reactor
414	D. Visser	Netherlands	CFD analyses of the ALFRED hot plenum
434	F. Lodi	Italy	The "ALFRED White Book": a business card of the project
501	S. Aithal	India	Design & Development of custom shaped back-up seal in silicone for PFBR

4.2. TRACK 2 – FAST REACTOR SAFETY

4.2.1. Session 2.1. General Safety Approach

Eight presentations were given in succession in Session 2.1 followed by an open discussion with all participants.

P. Calle Vives (IAEA) presented a summary of recent activities on reviewing the applicability of the IAEA safety standards to novel advanced reactors including fast reactors. She informed that the applicability review first relied on the identification of unique design features of the novel advanced reactors. Her presentation provided examples of distinct SFR and LFR barriers to fission product release in comparison to LWRs and summarized key aspects of the process for identification of areas of novelty related to fission product barriers. Her presentation concluded with a brief discussion on the path forward to complete the safety standards applicability review.

A. Alemberti (Italy) presented an overview of the LFR system safety characteristics as published in a recent Generation-IV International Forum (GIF) report. The presentation highlighted the main safety advantages and potential challenges of the LFR design tracks currently pursued under GIF framework and summarized the status of ongoing safety-related R&D activities. In performing LFR system safety assessments, the report placed emphasis on the fulfilment of the GIF safety and reliability goals, providing an outlook on further progress and safety-related R&D efforts needed to bring LFR systems to industrial deployment.

S. Kubo (Japan) presented a summary of collaborative efforts between CEA, France and JAEA, Japan in support of development of joint severe accident analysis and testing programme pursued in parallel with the ASTRID project. He outlined the two focus areas as joint development of the SIMMER code and extensive severe-accident testing programmes being pursued in Japan and France to support the validation efforts.

A. Moisseytsev (USA) presented an overview of safety analyses performed for the ARC-100 design to support its design and licensing. As an innovative 100 MW(e) SFR concept being developed by Advanced Reactor Concepts, LLC. in the USA, ARC-100 relies on inherent and passive safety, leveraging the safety characteristics of the metallic-fuelled fast spectrum core, pool-type configuration with passive redundant decay heat removal by DRACS and RVACS. To demonstrate its inherent safety potential, the presentation focused on the analysis of unprotected accidents assuming that the reactivity control systems fail to initiate scram, and the reactor power is controlled by reactivity feedbacks only.

G. Grasso (Italy) presented an overview of pre-licensing activities in support of building the Advanced Lead cooled Fast Reactor European Demonstrator (ALFRED) in Romania. He highlighted the benefits of this effort in facilitating early dialogue with the safety authority and contributing to mutual awareness of the novel technology. He explained that the ALFRED licensing framework involves defining the licensing rules and metrics, assessment of the applicability of the existing norms, guidelines, codes and standards, preliminary design evaluations to demonstrate that the safety approach meets the regulatory expectations, and establishment of the "Safety Demonstration Programme".

H. Yamano (Japan) provided an update on the Generation-IV International Forum (GIF) SFR Safety and Operation project activities that include:

- analyses and experiments that support establishment of the safety approaches and validate the performance of specific safety features,
- development and verification of computational tools and validation of models employed in safety assessment and facility licensing,
- acquisition of reactor operation technology, as determined largely from experience and testing in SFR plants such as Monju, Phenix, BN-600, EBR-II and CEFR.

J. Andrus (USA) presented an overview of the application of a risk-informed approach for design and authorization of the Versatile Test Reactor (VTR) in the USA. VTR is a 300 MW(th) metallic fuelled fast spectrum test reactor currently being developed with the mission to enable accelerated testing of advanced reactor fuels and materials. Mr. Andrus' presentation summarized how the guidance for the risk-informed approach (initially developed for commercial license applications by US NRC) was tailored for application to a DOE test reactor along with its implications and lessons learned.

P. Gauthe (France) presented a summary of Generation-IV International Forum (GIF) basis of safety approach for the diverse set of design and safety characteristics of all six Gen-IV systems. His presentation discussed the main ways to achieve the high-level safety objectives that include improved operational safety and reliability, reduced likelihood and degree of core damage, and elimination of the need for off-site emergency response. GIF basis of safety approach outlines a framework to achieve these overarching goals and assure that, during normal operation or anticipated transients, the safety margins are adequate, accidents are prevented, and off-normal situations do not deteriorate into severe plant conditions. These priorities continue the past trend and seek simpler designs that are intrinsically safe following the safety-by-design principles so that safety is “built-in” the fundamental design, rather than “added on” to the system architecture.

TABLE 7. PRESENTATIONS FROM SESSION 2.1 – GENERAL SAFETY APPROACH

Chairs: Tanju Sofu and Xiaoyan Yang supported by Anton Moisseytsev

ID	Presenter	Designating Member State/Organization	Title of the Paper
306	P. Gauthé	France	Basis for the Safety Approach (BSA) for design & assessment of Generation IV nuclear systems
538	P. Calle Vives	IAEA	Examples of areas of novelty in liquid metal fast reactors to consider in the review of applicability of the IAEA safety standards: fission product retention barriers: differences between liquid metal fast reactors and light water reactors
338	A. Alemberti	EC	System Safety Assessment of the Generation IV lead fast reactor
270	S. Kubo	Japan	France-Japan collaboration on the SFR severe accident Studies: Outcomes and future work program
241	T. Sumner	USA	Safety analysis of the ARC-100 Sodium-Cooled Fast Reactor
433	G. Grasso	Italy	Approach for ALFRED licensing in Romania
7	H. Yamano	Japan	Activities of the GIF safety and operation project of sodium-cooled fast reactor systems
268	J. Andrus	USA	Application of a risk-informed performance-based approach for the authorization of the Versatile Test Reactor

4.2.2. Session 2.2. Safety Design and Analysis

Session 2.2 comprised of 8 presentations, two from USA, three from France, two from India and one from Russian Federation.
T. Sumner (USA) presented an overview of the Versatile Test Reactor Safety Analysis. The analysis has been performed using the SAS4A/SASSYS-1 fast reactor safety analysis code with a model representing the reactor core, primary and intermediate heat transport systems, reactor vessel auxiliary cooling system, and reactor protection system. The main conclusions are as follows: primary heat transport system is able to transition quickly & effectively to natural circulation. Reactor Protection System dominates transient behaviours and inherent feedbacks. RVACS provides sufficient heat rejection capability. There are large margins for all criteria predicted for all analysed transients. VTR is being designed with a high level of safety.

J.-B. Droin (France) provided a presentation on integrating safety at the first design stages: a new methodology for safety-oriented SFR core design. He pointed out that an early consideration of safety in the design process enables an effective design review possibility toward intrinsically safe concepts and enables to get rid of a complex safety architecture. For this reason and several fast-running tools have been developed at CEA over the past years. Significant phenomena were identified and modelled for each boundary accidental sequence in order to build the severe accident fast running tool platform.

Three main initiators may lead to severe accident conditions in a SFR core: reactivity insertions, local subassembly faults and loss of the core cooling. In this report only the phenomenology related to the unprotected loss of flow (ULOF) is presented. The tools platform covers all kinds of accidental phenomenology, from several kinds of initiators (pump trip, reactivity insertion, local blockages, etc.) until reaching a stable and coolable reactor state. This makes possible the characterization of some major accident transient bifurcations in terms of probability of occurrence, of identification of the governing parameters or of consequences on the transient evolution.

S. Anurag (India) presented Thermal Hydraulic Assessment of Performance of SSDHRS. The talk was focused on secondary sodium decay heat removal system, which is envisaged for future reactors to meet the new safety norms of atomic energy regulatory body. Analysis of SSDHR System is carried out using system dynamics code Flownex. Flownex is a general-purpose one-dimensional code for thermal hydraulic simulation of systems with component level modelling capabilities. Heat transfer capacity of each SSDHR system is found to be 15.17 MW at 544°C temperature of hot pool sodium. Forced circulation flow rates of secondary sodium and air are estimated as 133 kg/s and 59.5 kg/s respectively. Parametric studies have been carried out by varying hot pool temperature in the range of 200°C to 650°C and primary sodium flow rate, and their effect on performance of SSDHRS is studied. Transient analysis of 'off-site power failure' event is carried out and the predicted hot pool and cold pool temperatures are found to be within the design safety limits. Further study is carried out to assess performance of SSDHRS during natural circulation of secondary sodium and air. Heat removal capacities at 544°C and 650°C primary sodium temperature are 8.78 MW and 11.07 MW respectively. Study is also carried out to assess the performance of SSDHRS, when all the three circuits are under natural circulation mode. The maximum heat removal capacity is found to be 9.775 MW when the hot pool temperature reaches 650°C.

A. Pantano (France) presented the pre-design of a passive decay heat removal system with a phase change material for SMR-SFR. The paper shows the main features of the pre-design of an innovative ex-vessel Decay Heat Removal System (DHRS) designed for a SMR-SFR of a core power of 400 MW(th). This safety system allows the heat removal in a fully passive way, only by the radiation of the outer surface of the primary vessel. The radiated heat is extracted by the DHRS through the enhanced natural circulation of a liquid metal flowing inside a bundle of tubes outside the vessel. The final heat sink of the DHRS is another important aspect of innovation. The aim is to diversify it, improving the safety function ensured by a standard water cooled or air cooled heat sink. For this purpose, the use of a Phase Change Material (PCM) has been chosen inside the heat sink in order to provide a supplementary thermal inertia, which contributes to improve the cooling capacity of the DHRS. The actual design of the ex-vessel DHRS has been applied to the new SFR sketch ATRIUM, in order to guarantee a fully passive cooling of the reactor core for a duration of three or more days.

J. Andrus (USA) made a presentation on the Development of the Versatile Test Reactor (VTR) Probabilistic Risk Assessment. First, he introduced the risk-informed authorisation process that was applied to the VTR, that was based on risk-informed performance based approach and VTR conceptual design PRA. He further introduced the SSC classification criteria used, which included both risk-informed and deterministic aspects. The results of the VTR conceptual design PRA were utilized to inform the categorization of safety basis events and classification of SSCs. They were also utilized to support multiple design analyses and comparisons. The Conceptual Safety Design Report also represents one of the first major safety documents utilizing the risk-informed performance based approach to be approved by the regulator body.

I. Shvetsov (Russian Federation) provided a presentation on the Analysis of the SGTR accident for safety justification of two-circuit lead cooled reactor. The overall goal of the presented analysis is justification for the fact that under steam ingress into the core conditions no safe operation limits specified for the reactor are exceeded. 3D multi-physics (neutronics + thermal hydraulics) UNICO-2F code was developed for studying SGTR accident. The code is capable of calculating transient 3D spatial distributions of coolant velocity, pressure and temperature, as well as steam concentration and power density in the core. The analysis of BREST-OD-300 reactor parameters under SGTR accident conditions was performed and it was shown that even for the most conservative scenario of the accident maximum (during the transient) fuel pin cladding temperature was kept within permissible limits. Therefore, the self-protection of BREST-OD-300 reactor in case of SGTR accident was confirmed.

Shri S. Raghupathy (India) presented the Design Studies Towards Raising Fast Breeder Test Reactor (FBTR) to Full Power. He introduced the FBTR and its core evolution. He noted that the power of FBTR has been raised to its rated power from 32 to 40 MW(th), with Mark-I subassemblies during the 30th campaign. The new core has 70 fuel SAs with the peak Linear Heat Rate (LHR) restricted to 400 W/cm. For ensuring a minimum shutdown margin of 4200 pcm, four poison subassemblies (10B enrichment 50%) are added in the second ring along with existing 6 control rod subassemblies (10B enrichment 90%) provided in the 4th ring. Various core design and safety studies have been carried out. In order to demonstrate the inherent safety characteristics and the capability of plant protection system with respect to various plant transients various plant dynamics studies have been carried out. The safety of the plant was demonstrated for the operation at 40 MW(th) power. Shielding analysis shows that in general, there is an increase in neutron and gamma fluxes at various locations of core and shield regions with respect to the 32 MW(th) core. The present shielding provided for the core & reactor assembly is verified to meet the safety requirements. Based on the limiting dpa of Grid Plate, FBTR is safe to operate for another 7.25 fpy at 40 MW(th).

T. Le Meute (France) presented Modelling of postulated reactivity insertion in a Generation IV Molten Salt Reactor. First, he introduced the context of the study and the MSR safety system under consideration. He noted that this concept can be operated in the Th/U cycle and a fluoride salt or in the U/Pu cycle and a chloride salt. The goal of this work is to study the MSFR behaviour in case of a postulated reactivity insertion. In order to evaluate the consequences of extremes reactivity insertions, the first study concentrates on slow reactivity insertions to verify the efficiency of the draining of the core. Because of the presence of the expansion tank, the beginning of the draining has no impact on the reactivity in the core. Then, it is needed to verify that there are wide safety margins between the consequences of plausible reactivity insertion and the consequences of extreme postulated reactivity insertions. In the case of extreme reactivity insertions, during the early stages of the transient, the salt cannot expand freely and goes out of the core. The pressure increases and the bubbles inside the salt collapse.
To make these studies, CEA is developing two independent modelling. The first one is being developed in order to study slow reactivity insertion. The second code aims at calculating fast explosion of the vapour formed in the salt in case of fast reactivity insertions.

TABLE 8. PRESENTATIONS FROM SESSION 2.2 – SAFETY DESIGN AND ANALYSIS

Chairs: Marina Demeshko and Hidemasa Yamano,

ID	Presenter	Designating Member State/Organization	Title of the Paper
240	T. Sumner	USA	Overview of the Versatile Test Reactor Safety Analysis
204	J.-B. Droin	France	Integrating safety at the first design stages: a new methodology for safety-oriented SFR core design
461	S. Anurag	India	Thermal Hydraulic Assessment of Performance of sodium system based decay heat removal circuit
123	A. Pantano	France	Predesign of a passive decay heat removal system with a phase change material for SMR-SFR
258	J. Andrus	USA	Development of the Versatile Test Reactor (VTR) Probabilistic Risk Assessment
376	I. Shvetsov	Russian Federation	Analysis of the SGTR accident for safety justification of two-circuit lead cooled reactor
488	S. Raghupathy	India	Design Studies Towards Raising FBTR to Full Power
152	T. Le Meute	France	Modelling of postulated reactivity insertion in a Generation IV Molten Salt Reactor

4.2.3. Session 2.3. Accident Analysis

The session comprised of eight presentations by experts from Belgium, India, Japan, the United States, Russia, and China on safety-related analyses of sodium cooled fast reactor (SFR), including those on source term, unprotected transients, and sodium fires. The presenters analysed quite mature design options and did show simulation results that were in general confirmed by experiments or by comparisons with other codes. The need for development of more advanced simulation tools and for further validation of existing ones was confirmed.

D. Petrovic (Belgium) presented a coupled neutronic and thermal hydraulic simulation of system behaviour of Fast Flux Test Facility (FFTF) during unprotected loss of flow (ULOF), focusing on a passive reactivity control device, the gas expansion module (GEM). The simulations of the LOFWOS test conducted during the operation of the FFTF showed that a simple Point Reactor Kinetics (PRK) model is appropriate as a modelling of neutronics of the reactor cores of the similar geometry, size and the fuel composition as FFTF. Furthermore, the mitigation performance of GEM for ULOF was shown to be similar to that of the sodium plenum of ASTRID, and thus effective in achieving inherent safety features. Possible next steps are to improve models for fuel and core thermal-mechanics behaviour.

P. Patel (India) presented a study on aerosol evolution in the containment of a 1250 MW(th) SFR after ULOF. Aerosol size distribution was represented by 150 bins from 1 nm to 0.01 mm. Different assumptions lead to different amounts of aerosol. After 24 hours less than ~2% of the total initial aerosol mass is suspended. The analyses show that Uranium and Plutonium remain mostly in the pool. Temperature and pressure in containment remain within design limits.

Y. Onoda (Japan) presented the results of the Work Package 2 (WP-2) in the IAEA Coordinated Research Project (CRP) aimed at evaluating fission product transport behaviour inside a reference pool-type SFR under severe accident conditions. In WP-2, where four institutions participated: NCEPU (China), IBRAE RAN (Russia), IGCAR (India), and JAEA (Japan), the mass of primary sodium ejected instantaneously in the reactor containment building was evaluated as a common benchmark problem. The results indicate that there is no significant difference in the accuracy of the analysis methods employed by the institutions for this WP. The need for analysis of the dynamic response of the plugs was suggested for the evaluation of sodium leakage into the top shield, taking into account transient sodium inflow into the leakage path between plugs.

J. Chang (USA) presented the results of WP-3 of the abovementioned IAEA CRP. WP-3 focuses on fission product behaviour in containment after ULOF. Seven organizations, participated in this study, in particular TP (USA), CIAE and XJTU (China), CEA (France), IBRAE (Russia), IGCAR (India), CIEMAT (Spain). To decouple this part of analysis from previous stages of calculation, the stand-alone calculation was defined for WP-3, which uses a set of pre-defined release fractions. In the coupled case, the release fractions of radionuclides computed at the previous work packages, in particular in WP-2, were used as initial conditions. The most challenging modelling task was to simulate complex phenomena of sodium pool fire, where different modelling assumptions were used, and different results were obtained. For the stand-alone cases, reasonable agreement between results of different models was obtained.

I. Pakhomov (Russia) presented a severe accident management (SAMG) strategy for removing heat from the reactor vessel by air circulation in the reactor vessel auxiliary cooling system (RVACS) during a severe accident in a high-power SFR. Analytical evaluation of decay heat removal characteristics by air circulation and radiation heat transfer from the reactor vessel showed that the proposed SAMG can keep the reactor vessel temperature below the limit even with natural circulation under the assumed accident conditions.

R. Sarangapani (India) presented the experience gained at IGCAR on radiological protection at the FBTR during 30 years of its operation. Several cases of radioactivity leaks were considered, and results of activity measurements were presented. In general, the obtained at FBTR experience provides a good basis for radiological protection at Indian SFRs in the future.

X. Jin (China) presented the safety analysis of a 300 MW(th) MOX-fuelled small module SFR (SMSFR) for anticipated transient without scram (ATWS) events. Based on the analysis of reactivity characteristics of SMSFR with the no-refuelling system, several measures for reactor design were proposed to mitigate the effects of ATWS and to ensure inherent safety performance during a long operating lifetime, in which the neutronic characteristics of the core vary considerably with changes in fuel burnup. It was suggested that over-deep insertion of control rods be avoided to control the excess reactivity of the SMSFR. There was a discussion on the validity of the natural circulation flow rate of coolant assumed in the analysis.

A. Moisseytsev (USA) presented a study on sodium fire hazards analysis and protection system methodology for Versatile Test Reactor (VTR). The presentation included main phenomena considered in the analyses and simulation models for sodium fires. Finally, an overview of the VTR sodium fire mitigation strategy was provided for each of the major areas of the facility where significant volumes of sodium would be located.

TABLE 9. PRESENTATIONS FROM SESSION 2.3 – ACCIDENT ANALYSIS

Chairs: Koji Morita and Andrei Rineiski

ID	Presenter	Designating Member State/Organization	Title of the Paper
194	D. Petrovic	Belgium	Coupled neutronic/thermal-hydraulic simulation of Unprotected Loss of Flow Test at Fast Flux Test Facility
68	P. Patel	India	Mechanistic modelling of aerosol evolution in an SFR containment following a hypothetical severe accident
18	Y. Onoda	Japan	Modelling and Simulation of Source Term for Sodium-Cooled Fast Reactor under Hypothetical Severe Accident: Primary System/Containment System Interface Source Term Estimation
277	J. E. Chang	USA	Modelling and Simulation of Source Term for Sodium-Cooled Fast Reactor under Hypothetical Severe Accident: Sodium Fire and Radionuclide Transport in Containment
59	I. Pakhomov	Russian Federation	The Severe Accident Management of the high-power SFR with loss of the heat removal from the core
509	R. Sarangapani	India	Over three decades of radiological protection experience at Fast Breeder Test Reactor (FBTR)
70	X. Jin	China	Safety Analysis of Small Modular Sodium Fast Reactors in Anticipated Transients Without Scram Scenarios
351	A. Moisseytsev	USA	The Versatile Test Reactor (VTR) Approach to Sodium Fire Hazards Analysis and Protection System Methodology

4.2.4. Session 2.4 Severe Accidents

Session 2.4 comprised eight presentations, one from Japan, one from India, one from Italy, two from Germany, and three from France.

A. Bachrata (France) presented the outcome of France-Japan Collaboration 2014-2019 during the ASTRID project. They tried to evaluate the consequence of postulated core disruptive accident (CDA) under hypothetical conditions where a serious power-to-flow mismatch is postulated. The selected scenario is ULOF transient without mitigation devices considering ASTRID core. The simulation resulted in the core molten pool hot spot reaching 5100 K, followed by upward ejection. The SIMMER-III code used in research was performed on both sides (French and Japanese teams) and post-processing focused on P-V work identification. The French team used EUROPLEXUS and the Japanese team used AUTODYN for benchmark, but no significant reactor vessel deformation was observed by either EUROPLEXUS or AUTODYN. Future work will focus on benchmark only between mechanistic codes followed by other scenario studies for chained calculation.

X-N. Chen (Germany) presented SIMMER ESFR-SMART Model in the aspects of geometric and thermal hydraulic model, neutronic feedback, thermal expansion. The thermal expansion model is included and the new CRDL model is developed and used. 4 ULOF cases with fuel/clad driven axial thermal expansion and two different steel thermal expansion coefficients for CRDL are calculated. Power excursions are obtained in the first 3 cases, with about 100 GJ thermal energy release. Long time calculation shows that the boiling oscillation can disappear, which suggests a higher initial cover gas pressure can prevent the sodium boiling here.

F. Payot (France) presented Severe Accident In-pile experiments for Generation IV reactors and Astrid project (SAIGA) programme, focused on safety demonstration robustness for innovative SFRs. SAIGA in-pile test consists of IGR reactor operated with the control rods, sodium loop and a test device for on-line measurements and post-irradiation examinations. A phenomenology of events of interest is consistent with the qualification of the Severe Accident calculation code. The date for the SAIGA test realization is planned for the beginning of 2025.

K. Kamiyama (Japan) presented the SFR severe accident studies that have been in progress using IGR in Kazakhstan, for which the JAEA is engaged in the EAGLE-3 project. The EAGLE programmes consist of the out-of-pile test and the in-pile test. Experimental studies for evaluating the effectiveness of design measures to achieve In-Vessel Retention have been carried out as a collaboration with the National Nuclear Center of the Republic of Kazakhstan. Experimental data to study dominant phenomena in the core disruptive accident were obtained through the EAGLE-1 and EAGLE-2 programmes and are being obtained in EAGLE-3. Through the validation of the safety analysis code SIMMER using data of EAGLE and CDA analyses by SIMMER, effectiveness of design measures against the post-accident material relocation and heat removal phases will be evaluated.

T. Sathiyasheela (India) presented the study result of uncontrolled withdrawal of control rods and in-pin fuel motion feedbacks. The UTOPY study consists of four cases in accordance with reactor conditions of uncontrolled withdrawal of control rods. In the case of Constant Inlet Coolant Temperature (CICT) and without In-Pin Fuel Motion (IPFM), Feedback Reactivity is 0 $. For another case with Time Dependent Inlet Coolant Temperature (TDICT) and without IPFM, Feedback Reactivity is also 0 $. With CICT and IPFM, Feedback Reactivity is -0.1 $ at 200 s and with TDCT and IPFM, Feedback Reactivity is about -0.113 $ at 200 s. The feedback from the in-pin-fuel motion is good contribution to the inherent safety of the reactor. Changes in the inlet coolant temperature have a considerable contribution in shaping the power profile and hence the temperature.

C. Journeau (France) introduced the status and future plan of France-Japan collaboration on fuel-coolant interactions of SFRs which are: Joint interpretation of steel-sodium interaction tests in MELF facility at JAEA; development of SERUA facility at CEA to study the film boiling heat transfer in sodium and preparation of a joint programme of experiments. The future new PLINE large-scale experimental platform to be funded by France Recovery plan is also a new path of collaborative work on severe accident experimental R&D.

S. Gianfelici (Italy) presented a result of transient 3D simulations for the Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) reactor. The preliminary results show interesting differences between 2D and 3D results, especially regarding the sodium boiling and rewetting behaviour, which greatly affects the reactivity swing and thus the power evolution during the ULOF transient. The results demonstrate the applicability of SIMMER-IV to full vessel SFRs for the simulation of ULOF initiation phase. This applicability was, until now, restricted by the very high computation costs and very long simulation times, and full vessel SFR models for transient analyses at Karlsruhe Institute of Technology (KIT) were mainly performed with the 2D SIMMER-III code. These limitations have been eased recently with modification and optimization of the code. The established model and code versions offer a basis for further SIMMER-IV code and 3D ASTRID model optimizations.

X. Gaus-Liu (Germany) presented the work on LIVE Core-Catcher (LIVE-CC). LIVE-CC experiments and TrioCFD protest simulation were performed to characterize the long-term heat transfer feature of an SFR core catcher. TrioCFD simulation proves its robust resolution and the rational calculation of temperature field. The heat removal at the upper surface was the main heat sink of a melt pool in the core catcher. TrioCFD precisely predicts the upper surface heat transfer, whereas it slightly overestimates the heat flux at the lateral surface and at the bottom. Progression in TrioCFD validation will be presented in ERMSAR2022.

TABLE 10. PRESENTATIONS FROM SESSION 2.4 – SEVERE ACCIDENTS

Chairs: Shigenobu Kubo and Frederic Serre			
ID	Presenter	Designating Member State/Organization	Title of the Paper
126	A. Bachrata	France	Development of methodology to evaluate mechanical consequences of vapor expansion in SFR severe accident transients: lessons learned from previous France-Japan collaboration and future objectives and milestones
395	X-N. Chen	Germany	Simulation of ULOF initiation phase in ESFR-SMART with SIMMER-III
155	F. Payot	France	The SAIGA in-pile experimental program to qualify the SIMMER calculation tool in SFR Severe Accident Conditions
22	K. Kamiyama	Japan	A Status of Experimental Program to Achieve In-Vessel Retention during Core Disruptive Accidents of Sodium-Cooled Fast Reactors
475	T. Sathiyasheela	India	Comparisons of Feedback under UTOPA with In Pin Fuel Motion Dynamics in Fast Reactors
302	C. Journeau	France	French-Japanese experimental collaboration on fuel-coolant interactions in sodium-cooled fast reactors
449	S. Gianfelici	Italy	Transient 3D simulations for the ASTRID reactor: preliminary results for the ULOF initiation phase
336	X. Gaus-Liu	Germany	Experiment and Numerical Simulations on SFR Core-catcher Safety Analysis after Relocation of Corium

4.2.5. Track 2: Poster Session

For Track 2, seven poster contributions were presented. Modelling of various safety related aspects in a sodium cooled fast reactor such as source term and radionuclide release modelling as well as shielding analysis for Versatile Test Reactors and probabilistic safety assessment of sodium fire were some of the topics covered in this session. Further discussions focused on thermal hydraulic simulation of Loss of Flow Without Scram Test in FFTF facility and novel method of using the zirconium foil to increase nuclear power plant hydrogen safety.

TABLE 11. POSTER PRESENTATIONS FROM TRACK 2

ID	Presenter	Designating Member State/Organization	Title of the Paper
69	P. Patel	India	Modelling of radionuclide release from primary system during a hypothetical severe accident in an SFR
174	J. Wang	China	Study on Sodium Fire PSA Methodology for Pool-Type Sodium cooled Fast Reactor
259	D. Grabaskas	United States of America	Development of the Simplified Radionuclide Transport (SRT) Code Version 2.0 for Versatile Test Reactor (VTR) Mechanistic Source Term Calculations
260	E. Orlova	Russian Federation	Increase of nuclear power plant hydrogen safety using zirconium accumulator
265	T. Fei	United States of America	Preliminary Shielding Analysis for the Versatile Test Reactor
391	S. Shahbazi	United States of America	Fast Reactor Source Term Modelling and Simulation Functional Requirements and Gap Assessment
462	V. Govindarajan	India	Thermal Hydraulic Simulation of Loss of Flow Without Scram Test in FFTF using DYANA-P code

4.3. TRACK 3 – FUELS, FUEL CYCLES, AND WASTE MANAGEMENT

4.3.1. Session 3.1. Fuel Cycle Scenarios

The session included four papers from Russian Federation, two delivered by French representatives (among which one was representing the Generation IV International Forum) and one from China covering approaches to closing the nuclear fuel cycle using fast reactors, insights on the European project PUMMA on Pu management, the fuel options for Generation-IV Sodium cooled Fast Reactors and aspects related to characterization of fast reactor fuels.

V. Dekusar (Russian Federation) from the State Scientific Center - Leypunsky Institute for Physics and Power Engineering, Joint-Stock Company (IPPE JSC) presented the initial stage of closing the NFC of two-component nuclear power: Challenges and solutions and talked about a study of different scenarios considered for closing the nuclear fuel cycle to select the most effective option for transition to a regime of a two-component nuclear power system with thermal and fast reactors. The planning of the initial stage of closing the nuclear fuel cycle has a key importance but there are challenges associated with technological and economic uncertainties. When transferring to two-component nuclear energy system, at the initial stage of closing the nuclear fuel cycle in Russia, the number of fast breeder reactors BN type fuelled with plutonium do exist to a small extent, and stocks of separated plutonium already are sizable and continue to grow due to pilot reprocessing spent nuclear fuel of VVERs. Therefore, for this period, it is proposed to abandon reprocessing the SNF from the BN reactors, for example to operate in the "open in plutonium" cycle with storage of this SNF. The results of scenario studies show that the amount of plutonium extracted from VVER SNF is sufficient for commissioning and continuing operation of a small series of BN reactors.

N. Chauvin (France) from the French Alternative Energies and Atomic Energy Commission (CEA) introduced the objectives and structure of the project covering all activities to be undertaken in the Work Packages in the Presentation of the new European project PUMMA devoted to plutonium management in the whole fuel cycle. The project is expected to demonstrate that fast reactors with the associated fuel cycle are crucial for a flexible and sustainable management of plutonium and will provide new results for improving the knowledge in all the steps of the fuel cycle while offering attractive opportunities to engage new generations of researchers.

E. Rodina (Russian Federation) from JSC "Proryv", Rosatom presented fuel cycle closure for high power fast neutron reactor and introduced some of the work conducted under the “Proryv” project, which is underway in Russian Federation that implies the development of a number of nuclear technologies of the new type large-scale nuclear power based on preferential use of fast neutron reactors operating in closed fuel cycle. Core designs of lead and sodium cooled fast reactors with mixed uranium-plutonium nitride fuel were modelled and the results indicated the possibility of operation of the various designs considered with equilibrium reactivity in the whole life cycle.

M. Xiao (China) from the China Nuclear Power Technology Research Institute Shenzhen, China presented perspectives and discussions on the modes and development path of China's commercial closed nuclear fuel cycle. China implements the established policy of closed nuclear fuel cycle for the sustainable development of nuclear power. However, a feasible development plan and roadmap to initiate and deploy a commercial closed fuel cycle in China is still needed. Based on international experiences and China’s situation, it is necessary to initiate a closed nuclear fuel cycle from mature commercial nuclear power plants in China as the initial stage of the closed fuel cycle to lay the foundation for the future advanced nuclear fuel cycle, analyse the initiating mode of China's commercial closed nuclear fuel cycle, review the nuclear fuel types to be utilized in the closed nuclear fuel cycle, and introduce the possible configuration and development path of China's closed fuel cycle in the future.

Y. Kotov (Russian Federation) from the National Research Center “Kurchatov Institute” presented the Potential Role of Fast Reactors with Heterogeneous Fuel Assembly in Development Nuclear Power Structure and highlighted the reasons why heterogenous fuel assemblies are a good choice as an intermediate solution before full transition to dense fuel in a two-component nuclear energy system. To assess the capabilities of a fast reactor with heterogeneous fuel assembly material balances of fissile nuclides were calculated for different scenarios. The scenario that was found most adequate for the transition to a two-component system is the one considering BN and VVER-TOI reactors running on MOX fuel.

F. Serre (GIF) from the French Alternative Energies and Atomic Energy Commission (CEA) and representing the Generation IV International Forum (GIF) presented Reference Fuel Options for Generation-IV Sodium cooled Fast Reactors and introduced the work conducted by GIF to evaluate the SFR advanced fuel options. The efforts undertaken entailed a comparison of the oxide, metal, and nitride fuel types with respect to fuel fabrication processes and fuel performances, to identify advanced fuel candidates for different applications. Additional R&D efforts were also focused on the minor actinide bearing fuels and high burnup capability evaluation. The roles of the fuel in safety cases, as well as the challenges associated with the fuel qualification were discussed. A variety of irradiation experiments for SFR oxide, metal, and nitride fuels were identified. Yet, fast spectrum irradiation capabilities are very limited internationally and fuel testing campaigns can require a great deal of time, effort and expense. Thus, advanced fuel performance modelling techniques simulating the fuel irradiation behaviour in the reactor may play a more significant role in future fuel qualification, with the main challenge being to validate the predictability of the complex fuel performance phenomena identified for each fuel type.

Y. Karazhelevskaia (Russian Federation) from the National Research Nuclear University MEPhI presented the influence of isotopic composition of plutonium in fast reactor fuel on the reactivity margin and introduced the analysis of the influence of the isotopic composition of plutonium on the reactivity excess of a fast reactor. Calculations were carried out for the BREST-OD-300 and RBETS-M reactors. Calculations have shown that the usage of a different composition of plutonium in the fuel assemblies manufacturing for fast reactors has a different effect on the reactivity excess, depending on the type of reactor.

TABLE 12. PRESENTATIONS FROM SESSION 3.1 – FUEL CYCLE SCENARIOS

Chairs: Amparo Gonzalez-Espartero (IAEA) and Alina Constantin (IAEA)

ID	Presenter	Designating Member State/Organization	Title of the Paper
28	V. Dekusar	Russian Federation	The initial stage of closing the NFC of two-component nuclear power. Challenges and solutions
361	N. Chauvin	France	Presentation of the new European project PUMMA devoted to Plutonium management in the whole fuel cycle
45	E. Rodina	Russian Federation	Fuel cycle closure for high power fast neutron reactor
382	M. Xiao	China	Perspectives and discussions on the modes and development path of China's commercial closed nuclear fuel cycle
156	Y. Kotov	Russian Federation	Potential Role of Fast Reactors with Heterogeneous Fuel Assembly in Development Nuclear Power Structure
406	F. Serre	GIF	Reference Fuel Options for Generation-IV Sodium-cooled Fast Reactors
205	Y. Karazhelevskaia	Russian Federation	The influence of isotopic composition of plutonium in fast reactor fuel on the reactivity margin

4.3.2. Session 3.2. Development of Innovative Fuels: Design and Properties Irradiation

Session 3.2 comprised of eight presentations, two from India, two from Japan, one from Hungary, one from USA, one from France, one from Russia.

S. Hirooka (Japan) presented work on Recent studies on fuel properties and irradiation behaviours of Am/Np-bearing MOX. Recent studies on the properties of mixed oxide (MOX) fuels are summarized by focusing on the influence of MA (Am and Np) addition. A major influence regarding fuel design is the decrease in thermal conductivity accompanied by the enhancement of impurity-phonon scattering and the increase in oxygen potential. The influence of MA addition on the fuel temperature during irradiation was evaluated using a simulation code developed in the Japan Atomic Energy Agency (JAEA). The addition of 5% Am and 5% Np in MOX pellets, which are 5.42 mm in diameter, increases by ~50 K at the centre of the fuel pellet at a linear heating power of 42.4 kW/m because of a decrease in thermal conductivity while the discrepancy vanishes as a larger central void is generated in MA-MOX. The results are critical in evaluating fuel performance and safety in using MA-MOX for reducing volume and toxicity of high-level radioactive wastes.

Z. Hózer (Hungary) presented work on Selection, testing and development of qualification procedure for ALLEGRO gas cooled fast reactor fuel and introduced the fuel types proposed for the new design of the ALLEGRO gas cooled fast on the basis of detailed review reactor. The first core will be built with MOX or UOX fuel in 15-15Ti stainless steel cladding. These fuel types have been widely used in different sodium cooled fast reactors. The second core of ALLEGRO will use refractory fuel. The primary candidate is carbide fuel – (UPu)C or UC – in SiC cladding.

T. Rajkumar (India) presented work on Design of metal fuel pin for test irradiation in FBTR &FOR future reactors and introduced the Indian programme that involves test irradiation of various binary and ternary metal fuels in Fast Breeder Test Reactor (FBTR). Towards this, test irradiation of sodium bonded metal fuel pins in FBTR core was proposed with various fuel compositions. For a typical composition (EU-23%Pu-6%Zr), the design details including the thermal and mechanical design of the pin are discussed in the paper. Also during transients, the maximum allowable flow reduction in the test Subassembly (SA) containing the metal fuel pins are discussed which helped in arriving at the blockage limits.

T. Segawa (Japan) presented Development of simplified fuel fabrication technologies for fast reactors. A simplified mixed oxide (MOX) pellet fabrication method is developed for the fabrication of high-density hollow MOX fuel pellets with a low oxygen-to-metal (O/M) ratio of less than 1.97 for fast reactors. The MOX pellets are fabricated through tumbling granulation, die wall lubrication pressing, sintering, and O/M ratio adjustment using the raw powder obtained from the microwave direct heating denitration method. Die wall lubrication pressing, sintering, and the O/M ratio adjustment of green pellets were performed. A pressing pressure of 400 MPa was suitable for obtaining sintered pellets with a density 95% of the theoretical density.

C.B. Jensen (USA) presented Advanced reactor experiments for sodium fast reactor fuels (ARES) project. Advanced Reactor Experiments for Sodium Fast Reactor Fuels (ARES) is a joint project between Idaho National Laboratory (INL) and the Japan Atomic Energy Agency (JAEA) to investigate the transient performance of metallic and mixed-oxide (MOX) fuels. Specifically, the project, guided by advanced modelling and simulation (M&S) tools, aims to experimentally evaluate the transient failure modes of high-burnup metallic and MOX fuels, as well as support the development and validation of M&S tools.

V. Blanc (France) presented Towards design guidelines for fast reactor oxide fuel pins with high Pu content: driving post irradiation examination by benchmarking European fuel performance codes. The framework of the European Commission call for proposal "horizon 2020", the project plutonium Management for More Agility, called PuMMA, is presented. This project started in October 2020 and will last four years. A work package is dedicated to the behaviour and safety of mixed oxide fuels with high plutonium content, which is essential for plutonium multirecycling or plutonium burning in fast reactors. This paper describes main goals, planning and status of this work package. Major task is the comparison of a large set of European fuel performance codes (FPC) based on three passed experimental irradiations of oxide fuel pins containing around 45% of plutonium: CAPRIX, irradiated in Phénix French Reactor, TRABANT1 and TRABANT2, irradiated in High Flux Reactor, HFR, in the Netherlands.

N.S. Dudala (India) presented work on Root cause analysis of FBTR failed fuel pin in Indian Fast Breeder Test Reactor (FBTR) with Mixed Carbide Fuel as the driver fuel since 1985. Mixed Carbide was chosen as the fuel due to its high stability with Pu rich fuel, compatibility with coolant and for its better thermal performance. Being a unique fuel of its kind without any irradiation data, it was decided to use the reactor itself as the test bed for this driver fuel. The fuel has performed extremely well, with the peak burn-up reaching 165 GWd/t. In the year 2011, MK-1 fuel SA that reached 148.3 GWd/t burnup in III ring of FBTR core had a single pin failure which was identified by both cover gas detectors as well as bulk Delayed Neutron Detectors (DND).

M. Krivov (Russia) presented work on Uranium and mixed uranium-plutonium nitrides thermal stability. The thermogravimetric method was used to study the behaviour of uranium nitride and mixed uranium-plutonium nitride (MNUP) in a helium flow and helium with nitrogen gas mixture at temperatures up to 1900°C. When heated in helium in the low-temperature range (<1500°C), a mass loss was found, which amounts to hundredths of a percent. In this case, mass loss occurs in 2 stages, accompanied by the release of nitrogen. It has been shown that sintered nitride fuel pellets may contain several percent of uranium sesquinitride U_2N_3, which decomposes in this range. Nitride fuel pellets were heated in a gas mixture of helium with nitrogen to study the formation of higher nitrides.

TABLE 13. PRESENTATIONS FROM SESSION 3.2 – DEVELOPMENT OF INNOVATIVE FUELS: DESIGN AND PROPERTIES IRRADIATION

Chairs: Nathalie Chauvin and Min Xiao			
ID	Presenter	Designating Member State/Organization	Title of the Paper
8	S. Hirooka	Japen	Recent studies on fuel properties and irradiation behaviours of Am/Np-bearing MOX
47	Z. Hózer	Hungary	Selection, testing and development of qualification procedure for ALLEGRO gas-cooled fast reactor fuel
503	T. Rajkumar	India	Design of metal fuel pin for test irradiation in FBTR &for future reactors
10	T. Segawa	Japan	Development of simplified fuel fabrication technologies for fast reactors
150	C.B. Jensen	USA	Advanced reactor experiments for sodium fast reactor fuels (ARES) project: Transient Irradiation Experiments for Metallic and MOX Fuels
399	V. Blanc	France	Towards design guidelines for fast reactor oxide fuel pins with high Pu content: driving post irradiation examination by benchmarking European fuel performance codes
504	N.S. Dudala	India	Root cause analysis of FBTR failed fuel pin
124	M. Krivov	Russian Federation	Uranium and mixed uranium-plutonium nitrides thermal stability

4.3.3. Session 3.3. Reprocessing, Partitioning, and Transmutation

This session included six papers in total: five from Russian Federation and one from India, covering transmutation aspects, partitioning and actinide burning using fast reactors, fuel fabrication and reprocessing for the BREST reactor and investigations on the electrolytic processes in the oxide-chloride melts.

E. Dzugkoeva (Russian Federation) from State Scientific Center - Leypunsky Institute for Physics and Power Engineering, Joint-Stock Company (IPPE JSC) presented work on the Feasibility study of heterogeneous transmutation of americium in fast reactors. Americium is considered as the most dangerous of the minor actinides and transmutation of external americium in the fuel of a fast reactor is possible when its content is high enough; at low content, americium will accumulate. One of the options investigated is the heterogeneous transmutation in the blanket in devices with a strong moderator (zirconium or yttrium hydride) that theoretically makes it possible to convert all loaded americium into fission products during one campaign, eliminate the need for multiple handling of it and its transmutation product - curium, and also eliminate the problem of high residual heat release. In this way, all americium formed in the fast reactor can be converted into fission products. An alternative method for the heterogeneous burning of americium is the use of irradiation devices with a strong moderator (zirconium hydride). This makes it possible to remove the problem of high residual heat generation in unloaded assemblies. For additional benefits, the burning in fast reactors of americium together with its predecessor ^{241}Pu, might be considered, due to the use of plutonium from spent nuclear fuel of thermal reactors with the minimum possible storage time.

A. Shadrin (Russian Federation) from JSC PRORYV presented work on Fabrication and reprocessing of mixed uranium-plutonium nitride fuel for reactor BREST and evaluated the results of the development of key technologies for closed nuclear fuel cycle fast reactors (with dense mixed uranium-plutonium nitride (MNUP) fuel). The results of MNUP fuel tests were presented, and options for producing MNUP fuel containing minor actinides were examined. The results of the development and validation of individual operations (fuel oxidation, extraction-crystallization refining of the undivided uranium-plutonium-neptunium mixture, production of actinide oxides by direct denitration, separation and separation of americium and curium) were reviewed. It was concluded that within the framework of the 'PRORYV' project, a prototype of the fast reactor fuel cycle is to be created.

A. Izhutov (Russian Federation) from RIAR JSC, Dimitrovgrad presented work on Heterogeneous burning of minor actinides (MA) in a fast reactor. The transmutation of minor actinides through irradiation in a reactor allows for improving the nuclear fuel efficiency due to energy produced by the MA fission, mitigating the problem of spent nuclear fuel (SNF) long-term activity as well as producing and extracting useful radionuclides. The homogeneous MA transmutation and heterogeneous burning of MA are feasible in a fast reactor. The concept of heterogeneous MA burning involves their inclusion into inert matrices to eliminate the secondary MA formation and their placement in specific pins in the FR core or blanket. Heterogeneous burning of MA in the FR blanket has a more flexible MA handling strategy and can be used to achieve high MA burning rates. In Russia, there is a unique opportunity for the MA transmutation in the existing fast reactors (BN-600, BN-800) and the overall study presented was focused on the developing the technology for heterogeneous burning of MA.

N. Desigan (India) from Indira Ghandi Center for Atomic Research, Department of Atomic Energy in Kalpakkham presented work on Advanced flow-sheet for partitioning of trivalent actinides from fast reactor high active waste. The single-cycle processing methods proposed for the separation of trivalent actinides from high-level liquid (HLLW) involves the separation of trivalent actinides and chemically similar lanthanides as a group, from HLLW followed by the mutual separation of lanthanides and actinides from the loaded organic phase using aqueous soluble complexing agents. The extraction behaviour of Am(III) and Eu(III) was studied in different potential solvent systems to evaluate the feasibility of using them for the single-cycle separation of trivalent actinides from high-level liquid waste.

S. Kviatkovskii (Russian Federation) from the Centre of Analytical R&D, Private Enterprise "Science and Innovations", ROSATOM State Atomic Energy Corporation presented multi-criteria comparison of the efficiency of minor actinides burning in different nuclear reactors based on the INPRO/IAEA KIND approach. He presented a comparison of the efficiency of minor actinides (MA) burning in various type of nuclear reactors. A set of criteria for comprehensive comparison of reactor technologies, based on the INPRO/IAEA KIND approach to multi-criteria assessment, has been prepared. This set of criteria includes indicators in such areas as the efficiency of MA burning, economics, safety, environment, readiness of reactor technology and infrastructure for its implementation. The evaluation and comparison procedure were carried out using the KIND-ET tool. It is shown that a comprehensive multicriteria analysis of various aspects of the technologies, as expected, led to estimates that differ from the approach in which technologies are compared only single criterion and without taking into account the influence of other equally important factors. And the cumulative assessment of technologies largely depends on the set development objectives. This means that each of the presented options can take the first place in the rating when certain priorities are selected.

A. Dedyukhin (Russian Federation) from Institute of High Temperature Electrochemistry, in Ekaterinburg presented work on Investigation of the anodic processes on the ceramic anode in the oxide-chloride melts. The conducted study demonstrated that the NiO-Li_2O ceramics is the inert anode material for the electrolysis of LiCl-KCl-Li_2O melts at the temperatures of 550-650°C. The electrode processes were studied by voltammetry. The reduction of uranium dioxide pellets by electrolysis using a ceramic anode resulted in a product containing 99.8 wt.% uranium metal. Thus, the NiO-Li_2O anode can be used for the spent nuclear fuel reprocessing in molten salts based on LiCl-Li_2O.

TABLE 14. PRESENTATIONS FROM SESSION 3.3 – REPROCESSING, PARTITIONING, AND TRANSMUTATION

Chairs: Amparo Gonzalez-Espartero (IAEA) and Akira Yamaguchi			
ID	Presenter	Designating Member State/Organization	Title of the Paper
61	E. Dzugkoeva	Russian Federation	Feasibility study of heterogeneous transmutation of americium in fast reactors
63	A. Shadrin	Russian Federation	Fabrication and reprocessing of mixed uranium-plutonium nitride fuel for reactor BREST
119	A. Izhutov	Russian Federation	Heterogeneous burning of minor actinides in a fast reactor
493	N. Desigan	India	Advanced flow-sheet for partitioning of trivalent actinides from fast reactor high active waste
134	S. Kviatkovskii	Russian Federation	Multi-criteria comparison of the efficiency of minor actinides burning in different nuclear reactors based on the INPRO/IAEA KIND approach
544	A. Dedyukhin,	Russian Federation	Investigation of the anodic processes on the ceramic anode in the oxide-chloride melts

4.3.4. Session 3.4. Advanced Fuel Development

Session 3.4 comprised of seven presentations, three from Russian Federation, one from Belarus, one from Egypt, one from Pakistan and one from USA.

A. Gulevich (Russian Federation) presented on the investigations for changing the isotopic composition of plutonium from the spent MOX fuel of PWRs in fast reactors. It examines large-scale Pu improvement in BN- type fast reactors with increased conversion ratio. The measurable parameters for an experiment in an operating BN-800 FR were also provided to demonstrate multi-recycling with change in isotopic compositions of plutonium and may also lead to a new method of nuclear material accounting. The investigation thus far has looked only at recycle of PWR fuel into fast reactors, but further recycle for fast reactors back to PWRs will also be investigated.

L. Capriotti (USA) presented the study of post irradiation characterisation of advanced fuel campaign (AFC) metallic fuel (U-10Zr/U-20Pu-10Zr, U-TRU-Zr, and other variants), sodium bonded and irradiated in the Advanced Test Reactor (ATR). The investigation assessed swelling rates and fuel cladding chemical interaction (FCCI) and impacts to fuel performance of additions of minor actinides (Am,Np). The PIE study on behaviour of an innovative metal fuel (U-Zr with Pd addition) at low burnups (between 2-4%FIMA) from various irradiation experiments in ATR described indicate that Pd addition can help mitigate FCCI, but additional Zr in the alloy might be necessary to ensure sufficient Zr remains in solid solution with the U. Overall, the fuel has performed acceptably well and does not get affected by MA addition.

A. Belyaeva (Russian Federation) presented the results of post irradiation examination of mixed nitride fuel pins (SNUP) for BN-1200 and lead cooled BREST OD-300. The fuel pins with SNUP with gas (helium) irradiated up to 7.5, 6.0 and 4.5 at% burnup, in BN-600 and BOR-60 for 10 EFA were studied. The PIEs revealed the swelling rates decrease as burnup is increased. Furthermore, swelling rate is lower for lead-bonded fuel than for helium-bonded fuel, likely due to the lower fuel temperature of the lead-bonded design. The cladding mechanical tests results show a significant margin of strength and plasticity at normal operating temperatures. The results also confirmed the performance of mixed nitride pins at the test parameters and the possibility of their irradiation prolongation to higher fuel burnup.

A.B. Maqbool (Pakistan) presented an overview of status of various types of nuclear fuels for fast reactors. It provides a comparison of properties and technical issues of metallic fuels, nitride fuels, oxide fuels dispersion fuels and special mechanical fuel forms (particle fuels, sphere-pac and vibro-pac). The author indicated that Pakistan is not pursuing a specific long-term fuel cycle technology at this time but is conducting investigations such as this to assess available technologies.

A. Bakhin (Belarus) presented the results of the reactor experiment of uranium-zirconium carbonitride fuel carried out in the SM-3 reactor at the JSC "SSC RIAR. The LEU (19.75% by U-235) UZrCN fuel investigation reported is for irradiation to a burnup of 0.63%FIFA, but the intention is to continue the investigation to burnup of 40%. The PIE investigations (gamma scanning of the experimental capsule) demonstrated that the fuel column did not have ruptures and the length of the fuel column, after irradiation did not change. The conducted material testing demonstrated that the additional fuel caking (or in-reactor sintering) resulted in the decrease of the fuel tablets (in height and diameter) during irradiation by 1% approximately, whereas the density increased by 3% approximately.

A. Komarov (Russian Federation) presented an investigation of the potential radiological concerns of chemical forms of fission products in mixed nitride fuels. Chemical compounds of ruthenium, rhodium, palladium; strontium, yttrium and cerium nitrides and lanthanide nitrides are formed directly in the fuel matrix (irradiated nitride fuel) during the operation of reactors or are possibly formed during the processing of irradiated nuclear fuel. Consideration of the results provides insight into their physicochemical and radiotoxicological properties to assess radio-hygienic hazard.

M.A. Ibrahim (Egypt) presented two computational models, with homogeneous and heterogeneous modelling of fuel assembly design features, for the large-scale Gas Cooled Fast concept GFR2400 using Monte Carlo transport code, MCNPX, to study and analyse the neutronic behaviour and transmutations capabilities of the reactor core for fuel burnup of 1443 EFPD. The results indicate 99% breeding ratio without the use of fertile blankets. The computational results for homogeneous and heterogeneous models were nearly identical, due to the relatively long neutron mean free path in the fast spectrum, suggesting that the simpler homogeneous model is likely to be sufficient by many fuel cycle evaluations.

TABLE 15. PRESENTATIONS FROM SESSION 3.4 – ADVANCED FUEL DEVELOPMENT

Chairs: Doug Crawford and Kailash Agarwal (IAEA)			
ID	Presenter	Designating Member State/Organization	Title of the Paper
143	A.Gulevich	Russian Federation	On the possibility to change the isotopic composition of plutonium from the spent MOX fuel of PWRS in fast reactors
164	L.Capriotti	USA	Postirradiation characterization of AFC metallic fuel alloys concepts
213	A.Belyaeva	Russian Federation	Results of post-irradiations examinations of mixed nitride pins with gas and liquid metal sub-layers
412	A.B.Maqbool	Pakistan	Nuclear Fuels for Fast Reactors-A Review
319	A.Bakhin	Belarus	Low enrichment nuclear fuel based on uranium-zirconium carbonitride: reactor tests and preparation for studies at critical assemblies
239	A.Komarov	Russian Federation	Types of chemical compounds in the assessment of radiation and hygienic hazards when working with irradiated nitride fuel
11	M.A. Ibrahim	Egypt	Analysis of Fuel Burnup and Safety Parameters of Gas Cooled Fast Breeder Reactors

4.3.5. Track 3: Poster Session

The Poster session for Track 3 included 13 contributions. It focused on fusion fuel cycle, transmutation of minor actinides, optimization of PUREX process, features of high-level liquid waste storage tanks, development of components for application in reprocessing plants and use of Artificial Intelligence to predict process related failure in a reprocessing plant. Further contributions involving the topics such as reprocessing of nitride and metallic spent nuclear fuel using molten salts, electrical conductivity of multicomponent chloride melts and electrolytic reduction of the simulated oxide spent nuclear fuel were also presented.

TABLE 16. POSTER PRESENTATIONS FROM TRACK 3

ID	Presenter	Designating Member State/Organization	Title of the Paper
9	M. Nishina	Japan	Development of density control technologies for MOX pellet using dry recycled powders
157	E. Kulikov	Russian Federation	Controlled thermonuclear fusion: potential role of a joint (Th-U-Pu) nuclear fuel cycle
162	A. Savelev	Russian Federation	Revealing the dependencies of partitioning americium-241 and uranium using sorption technology based on solid-phase extractant TODGA
168	A. Terekhova	Russian Federation	Transmutation of minor actinides in a fast reactor with uranium-curium fuel
481	A. Krishnamurthy	India	Optimization of Ruthenium concentration in PUREX Process during Fast reactor fuel Reprocessing
484	P. Sivakumar	India	Assay of Waste drum based on Passive Neutron Counting Technique
490	K. Dhananjeya	India	Design, manufacturing and transportation of high capacity High Level Liquid Waste Storage tanks
491	L. Falix	India	Evaluation of EPDM and Silicone rubber compounds for application in Reprocessing Plant
495	S. Kumar R.V.	India	Development of Artificial Intelligence through PLC & SCADA to predict process related failure and abnormality in a Reprocessing Plant
541	A. Potapov	Russian Federation	Reprocessing of nitride and metallic spent nuclear fuel using molten salts
543	A. Potapov	Russian Federation	Electrical conductivity of multicomponent chloride melts, containing ions of mono-, di-, and trivalent metals
545	Y. Zaikov	Russian Federation	Determination of the metallic and oxide compounds in models based on metallic uranium containing uranium dioxide, metallic neodymium, cerium as well as neodymium and cerium oxides

TABLE 16. POSTER PRESENTATIONS FROM TRACK 3 (CONT.)

ID	Presenter	Designating Member State/Organization	Title of the Paper
546	A. Dedyukhin	Russian Federation	Electrolytic reduction of the simulated oxide spent nuclear fuel in LiCl-Li2O melt

4.4. TRACK 4 – FAST REACTOR MATERIALS (COOLANTS, STRUCTURES) AND COMPONENTS

4.4.1. Session 4.1. Advanced Reactor Cladding and Core Material, Coolants, and Related Chemistry

Session 4.1 comprised eight presentations, one from China, one from France, two from India, two from Russian Federation, and two from the United States of America.

P. Reddy G.V. (India) presented the overview of studies dedicated to creep and tensile properties of Indian advanced Fast Reactor Clad tubes (IFAC-1 for future FBRs). This paper introduced the details of the development of IFAC-1 and compared its properties with similar material developed elsewhere in the world, namely, Valinox D9 SS (France). The strength of the former had been found to be higher in the range 300-923 K. Further it emphasized that this new material exhibits promising creep rupture strength in a wide range of stress, viz., from 100 to 255 MPa. This could be a promising clad material suitable for burn ups up to 150 GWd/t.

M. Orlov (Russian Federation) presented the national studies on the heat conducting liquid metal sublayer, i.e. metal bonded U,Pu, mixed nitride fuel elements and compared its performance with that of those with a He sublayer, i.e. with He bonding. The advantages of the metal bonding thanks to the higher thermal conductivity of the Pb-Bi alloy which helps increase the fuel clad gap and provides enough room for free swelling of the nitride fuel. This in turn helps improve the fuel performance and lessens the severity of the fuel clad mechanical interaction (FCMI). This paper also discussed the formation of a protective zirconium nitride layer on the stainless steel clad. This is brought about when Zr is added as minor alloying component into the liquid metal used as the bond. The zirconium nitride acts as a barrier layer and prevents further diffusion of the corrosive fission products in to the clad.

J. Li (China) presented the studies carried out in China on the fabrication and performance assessment of 14 Cr FeCrAl cladding tube material. This belongs to the oxide dispersed steels (ODS) which had been cold rolled. They optimized the process of cold rolling. This had been carried out through the microstructural examination of the cold-rolled alloy. These authors also demonstrated the homogeneous distribution of the zirconia nano particles in the alloy matrix after cold rolling. However, the final product was found to possess a gamma fibre texture along the <111>//ND orientation. These authors claimed that this preferential orientation may not influence the mechanical properties of the final product significantly both axially and along its circumference.

E. Kent (USA) presented Gear Test Assembly (GTA) which is an experimental apparatus designed to test mechanical components, specifically gears and bearings, used in advanced fuel handling systems of liquid-sodium cooled fast-spectrum nuclear reactors. The existing data are insufficient for proper lifetime calculations of gearing components which operate under load in a high temperature liquid-sodium environment. A total of 11,184 simulated fuel assembly manoeuvres (372.8 hours of continuous use) have been completed to date using the original set of Inconel 718 gears. Testing has twice been paused to replace failed mechanical bearings. The performance of 52100 steel tapered roller bearings, with and without heat treatment, has been investigated. Extensive pre- and post-op non-destructive evaluation has been performed to record the health of the gears over the course of operation. Eddy current testing methods were used for detection of surface and subsurface flaws on the gear teeth. Extensive temperature/vibration/torque/sodium-oxygen content data have been collected by the GTA during two experimental campaigns. The results were very impressive, and they will be completed by additional tests. During discussion, it was confirmed that the duration of 20 000 hours is a rather conservative domain to qualify the gears.

R. Vijay Kumar (India) presented the work carried out in India on the irradiation induced damage in both SS 316 (LN) and SS 304 (LN). This paper highlighted tests carried out on different specimens that were test irradiated in the Fast Breeder Test Reactor, Kalpakkam, India. The objective of their study was to compare the irradiation performance of these two materials to explore the possibility of replacing SS316 (LN) which is currently being used in FBTR for different structural components with SS304 (LN) for applications that warrant its use below 400°C. These investigations were carried out on the tensile specimens irradiated in a special purpose test subassembly in the reactor. Even though both these materials possess a ductility limit well above the desired 10% limit, still SS 316(LN) would be the material of choice for it possesses better tensile and impact properties as compared to AISI 304(LN).

C. Latgé (France) firstly underlined the large amount of sodium uses for various purposes: chemistry, basic research, nuclear and solar energy then presented a review of the processes developed to support the safe and reliable operation of sodium systems. Processes dedicated to cleaning and decontamination, sodium treatment were described, showing the great benefits of the Na-water interaction, when it is well mastered. The use of alcohols to treat residual sodium has been prohibited, due to induced hazards linked to their use. In all sodium systems, it is necessary to control the sodium quality with regards the two main impurities oxygen and hydrogen, to avoid potential deleterious effects, including corrosion, plugging of narrow sections, loss of heat transfer efficiency in heat exchangers. Operational feedback has shown that cold trapping is the best process to control the quality. Several codes have been developed by CEA to support design studies among them OSCAR-Na to simulate activated corrosion products mass transfer, KUTIM for tritium mass transfer and more recently ANAÏS code for cold traps design and operation monitoring. Most of the Na processes can be adapted to various non-nuclear applications using sodium, without significant changes.

V. Alekseev (Russia) presented a review of investigations done for sodium purification. The scientific justification was developed for the design of BN-350, BOR-60 and BN-600 cold traps. To improve the safety of a new fast neutron reactor, the BN-1200 project, it has been decided to place the purification system in the reactor primary vessel, as it was done for Superphenix, in France. Based on calculations, using the developed codes for modelling thermal hydraulic and mass transfer processes in cold trap, optimal technical solutions are investigated. Calculations using the developed computational codes (TURBOFLOW, MASKA-LM and OpenFOAM) including thermal hydraulics and mass transfer processes in a cold trap, allowed to propose new solutions to improve the design of integrated cold trap in order to increase its efficiency and loading capacity for impurities. For integrated in-vessel sodium purification system getter traps are an alternative to cold traps. To purify sodium from oxygen, both insoluble and soluble getters were used. Tests of sodium purification with zirconium getter were carried out at temperature of 550°C. As soluble getter, granular magnesium was tested also, at 300°C.

H.T. Chien (USA) presented studies dedicated to the development of diffusion-type hydrogen meters (DTHMs), used to detect water ingress in the secondary sodium. Among all the steam generator (SG) leak detection techniques, this one is highly sensitive and effective in detecting small leaks to prevent further tube failure due to wastage propagation. Argonne has developed two types of DTHM, cover-gas hydrogen meter (CGHM) and in-sodium hydrogen meter (ISHM), to be integrated into the steam generator leak detection system of an SFR. Design specifications and operating conditions of the two DTHMs were identified. CGHM and compact ISHM were designed for fast response, reduced cost, simplicity, sensitivity, and seismic ruggedness. A test apparatus for performance evaluation of CGHM prototypes was designed and constructed. CGHM prototypes were tested with different hydrogen-argon mixtures in dynamic and equilibrium modes. Both modes demonstrated that hydrogen detection sensitivity of ISHM was down to 2 ppm or less in argon cover gas as required by the specification. Under the dynamic mode, the CGHM prototype has a response time around 1 sec. The equilibrium shows better consistency but with longer response time. Compact ISHM is operating without a sodium pump and is mechanically supported by the sodium line to which it is closely coupled. Both DTHMs, if calibrated, can provide real-time, direct measurements of hydrogen concentration or pressure. These tests have clearly shown that it is always possible to optimize the detection of Na-water interaction, for future SFRs.

TABLE 17. PRESENTATIONS FROM SESSION 4.1 – ADVANCED REACTOR CLADDING AND CORE MATERIAL, COOLANTS, AND RELATED CHEMISTRY

Chairs: Christian Latgé and K. Ananthasivan

ID	Presenter	Designating Member State/Organization	Title of the Paper
317	Prasad Reddy G.V.	India	Creep and Tensile Properties of Indian Advanced Fast Reactor Clad tubes (IFAC-1) for Future FBRs
254	M. Orlov	Russian Federation	Thermally conductive liquid-metal sublayer in fuel element
175	J. Li	China	Fabrication and performance assessment of ODS fecral cladding tube
252	E. Kent	USA	Gear Test Assembly: First Liquid Metal Component Testing in METL
322	R. V. Kumar	India	Influence of Low Dose Irradiation on Permanent Core Structural Materials of PFBR
537	C. Latgé	France	Sodium coolant: interaction with its environment and coolant processing
41	V. Alekseev	Russian Federation	Investigation of sodium purification
367	H.T. Chien	USA	Development and Demonstration of Diffusion-type Hydrogen Meters for Sodium-cooled Fast Reactors

4.4.2. Session 4.2. Structural, Novel, and Large Component Materials

This session consisted of six presentations. One from Saudi Arabia, one from India, one from Japan, two from China and one from the European Commission.

A. Alomari (Saudi Arabia) delivered a presentation on creep and creep-fatigue behaviour of the Advanced Stainless Steel Alloy 709 (Fe-25wt.%Ni-20%Cr). Alloy 709 is of current interest for structural applications in the Sodium cooled Fast Reactors owing to its desired set of properties including mechanical properties relative to conventional austenitic stainless steels. The microstructural observations of the crept specimens suggest that high-temperature dislocation climb deformation is the rate-controlling creep mechanism in the alloy. Additionally, Larson-Miller Parameter (LMP) and Monkman–Grant relationships were developed using the creep rupture data.

R. Mythili (India) delivered a presentation on the development of plasma nitriding treatments for the realization of hard deposits for enhancing the wear and fatigue resistance of large size and intricate fast breeder reactor components. The authors report on the realization of the plasma nitriding surface treatment for the casing ring of the secondary sodium pump of PFBR and the assessment of their performance in static sodium.

K. Toyota (Japan) presented an overview on the Material Data Acquisition Activities carried out by the Japan Atomic Energy Agency (JAEA) to develop the Material Strength Standard for Sodium cooled Fast Reactors for the revision of the standards of the Japan Society of Mechanical Engineers (JSME). The material tests conducted by the JAEA for evaluating the long-term creep properties, the material tests for the development of high-cycle fatigue evaluation methods, and the material tests for the development of ultra-high temperature material strength standards were reported as well.

S. Wei (China) presented a paper on the δ-ferrite transformation behaviour and mechanical properties of 316H weld metal during high temperature service. The 316 weld metals with different carbon contents were aged at 550°C and 600°C for different times and the microstructural evolution as well as the mechanical properties were evaluated and analysed.

R. Novotny (European Commission) presented a talk on the experimental campaign carried out on tensile testing of sub-sized specimens of 316L base steel and welds in liquid lead. After a description of the experimental apparatus, the results of slow-strain rate tensile tests of 316L(N) and its SAW welded joints in oxygen-controlled liquid lead environment are reported. In the experimental conditions considered, the results did not reveal any indications for the existence of liquid metal embrittlement and/or environmental-assisted cracking of 316L(N) and welds in liquid lead.

B. Xie (China) reported on a novel method of manufacturing of large reactor components based on metal Additive Forging technology. The realization of the heavy integral prototype support ring (φ=15.6 m) by means of this method is described as well as the results of the mechanical and microstructural characterizations. The test results indicate that the microstructure and mechanical properties of the support ring in different directions and regions are uniform and stable and meet the design requirements so confirming the reliability of this technology.

TABLE 18. PRESENTATIONS FROM SESSION 4.2 – STRUCTURAL, NOVEL, AND LARGE COMPONENTS MATERIALS

Chairs: M. Angiolini (Italy), E. Orlova, (Russia)			
ID	Presenter	Designating Member State/Organization	Title of the Paper
39	A. Alomari	Saudi Arabia	Creep and Creep-Fatigue Behaviour of an Advanced Stainless Steel (Alloy 709) - Application to Sodium-Cooled Fast Reactors
332	R. Mythili	India	Development of Plasma Nitriding as alternate hardfacing technique for Large components of FBR and Assessment of static In-Sodium Stability of Plasma Nitrided Layer
106	K. Toyota	Japan	Material Data Acquisition Activities to Develop the Material Strength Standard for Sodium-cooled Fast Reactors
178	S. Wei	China	The δ-ferrite transformation behaviour and mechanical properties of 316H weld metal during high temperature service
510	R. Novotny	EC	Tensile testing of sub-sized T91 and 316L steel specimens in liquid lead
80	B. Xie	China	A novel method of manufacturing a heavy integrated support ring in fast reactor

4.4.3. Track 4: Poster Session

Seven contributions were accepted for poster presentation in Track 4. Novel information on the topic of structural, novel, and large components materials were presented, including measures to prepare the heavy liquid metal coolant for filling the primary circuit of the reactor facility such as coolant purification, ensuring corrosion resistance of structural steels and cleanup of the cover gas. Further, results of investigation of the state of two types of highly enriched B_4C after irradiation in emergency protection pins were reported. The influence of preheating temperature mechanical properties of 12%Cr steel weld metals, material which is used in primary core and the possibility of oxygen concentration measurement in sodium were discussed. Further papers focused on structural materials such as investigation study of sodium interaction with thermal insulation and effects of reaction products on the structural material and impact of core materials on the cladding irradiation damage and impact of cladding material on neutronic balance in Breed-and-Burn fast reactors; concept, in which depleted or unenriched fuel is loaded in a core and breeds fissile fuels while burning them.

TABLE 19. POSTER PRESENTATIONS FROM TRACK 4

ID	Presenter	Designating Member State/Organization	Title of the Paper
57	A. Legkikh	Russian Federation	Current state and issues of the heavy liquid metal coolant technology development (Pb, Pb-Bi)
66	E. Kinev	Russian Federation	The working capacity analysis of boron carbide after two-year operation as an emergency protection material of the fast reactor
82	N. Loginov	Russian Federation	On measurement of oxygen concentration in sodium by means of plug indicator
236	D. Wu	China	Influence of preheating temperature on delta-ferrite formation and mechanical properties of 12%Cr steel weld metals
393	O. Sambuu	Mongolia	Impact of Core Materials on The Fuel cladding Irradiation Damage in Breed-and-Burn Fast Reactors
397	T. Obara	Japan	Impact of Cladding Material on Neutronic Balance in Breed-and-Burn fast reactors
445	Ch. Avinash	India	Experimental study on sodium insulation interaction and its effect on structural material

4.5. TRACK 5 – TEST FACILITIES AND EXPERIMENTS

4.5.1. Session 5.1. Experimental Reactors and Facilities

Session 5.1 comprised nine presentations: four from Russia, four from USA, and one from France.

J. Kuzina (Russian Federation) introduced diverse test facilities designed in Russian Federation. For example, "6B", liquid metal test facility for thermal hydraulic studies, is designed to study hydrodynamics, heat transfer and temperature regimes in the elements of cores, "metal-metal" heat exchangers, and mixing chambers of fast reactors under nominal and transient conditions. "AR-1", high temperature liquid-metal test facility, is designed to carry out research into thermohydraulic processes under emergency conditions of sodium fast reactor operation, including severe accident progression with sodium boiling in the core. "SPRUT", multipurpose liquid metal test facility, is designed to study thermal hydraulic characteristics of steam generators in reactors with liquid metal coolants. "Protva-1", test facility, is designed to solve the issues of sodium coolant technology as applied to fast reactor, including such tasks as to study mass transfer of impurities in a circulation loop with sodium, to study the improved methods for sodium purification from impurities, and to test the devices that control impurities in the coolant.

S. Pavlovich (Russian Federation) presented the study of modelling the thermophysical processes with liquid metal coolants. Experimental thermophysical studies carried out using the theory of physical modelling of hydrodynamics and heat transfer in liquid metals made it possible to obtain data not only on physics, but also on the characteristics of thermophysical processes in the fuel assemblies of the core and equipment units of reactors with liquid metal cooling. These data are the basis for the development of both approximate engineering methods for the thermal hydraulic calculation and precision methods, and three-dimensional thermal hydraulic codes.

M. Weathered (USA) presented the overview of a sodium fast reactor thermal hydraulic test facility. Thermal Hydraulic Experimental Test Article (THETA) system is an experimental facility to provide validation data and to develop thermal hydraulic components and instrumentation for liquid sodium. THETA has two purposes; one is system code and CFD validation, while the second is liquid metal thermal hydraulic component development and control strategies.

J. M. Such (France) presented an overview of R&D on sodium spray fire and sodium pool fire consequences. A great number of results from R&D programmes consist of experimental data, good knowledge in the field of sodium fires including implemented in computer codes, know-how to design and perform experimentation and several test devices patented. Results supported safety assessments of French SFRs including PHENIX and SUPERPHENIX reactors.

Ch. Grandy (USA) presented the overview of the Mechanisms Engineering Test Loop (METL). METL is a facility to test small or intermediate scale advanced liquid metal components and instrumentation in sodium. METL has had a long operational history for over 3.5 years including a drain and freeze for eight months outage followed by a throw and refill. They have developed and tested both component-type and thermal hydraulics-type experiments. Next test articles are full-scale test article "GTA" and a flow sensor test article "THETA".

D. Klinov (Russian Federation) presented an overview of the research reactor facility Multi-Purpose Fast Neutron Research Reactor (MBIR). The MBIR reactor is required for implementation of the strategy of the two-component nuclear power in Russia for the period up to 2050 based on fast and thermal reactors. The main purpose of the research reactor facility MBIR is to conduct extensive reactor tests of innovative materials and core component for Gen IV. The potential experimental capabilities of the reactor are sufficient for conducting research under the national nuclear industry development programme s and for fulfilling works by foreign customer, including joint international projects within the framework of the MBIR-based international research centre.

J. Roglans-Ribas (USA) presented a status of the Versatile Test Reactor (VTR) project. The goal of the VTR facility is to support the advanced reactor development and enable long-term innovation. In 2021, the risk reduction studies were performed, and industrial design & construction team was selected. The National Environmental Policy Act (NEPA) process is set to be completed in late 2022.

A. Zagornov (Russian Federation) presented an overview of the multipurpose fast research facility MBIR. MBIR is the world's largest fast neutron research reactor which is being constructed by Rosatom. It is planned to start experiments in 2028. It will support national scientific research programme for advanced experimental research for innovative technologies and best management practices.

F. Heidet (USA) delivered an update on the VTR, a 300 MW(th) pool-type SFR, offering fast flux above 60 dpa/year in large volumes including accelerated fuel and material irradiation testing in a variety of coolants. Core design is in the preliminary design phase, with all major aspects already designed and assessed. Next steps involve thorough uncertainty quantification in support of licensing and fine design adjustments.

TABLE 20. PRESENTATIONS FROM SESSION 5.1 – EXPERIMENTAL REACTORS AND FACILITIES

Chairs: Jordi Roglans-Ribas and Hideki Kamide			
ID	Presenter	Designating Member State/Organization	Title of the Paper
224	J. Kuzina	Russian Federation	Complex of experimental facilities for design and safety justification of fast reactors with liquid metal coolants
206	S. Pavlovich	Russian Federation	Physical modelling of hydrodynamics and heat exchange in fast reactors with liquid metal coolants
238	M. Weathered	USA	Overview of a Sodium Fast Reactor Thermal Hydraulic Test Facility
435	J.-M. Such	France	Overview of the R&D programs led by the past at IRSN on sodium fire
340	Ch. Grandy	USA	Mechanisms Engineering Test Loop (METL) Facility
89	D. Klinov	Russian Federation	Experimental capabilities of the research reactor facility MBIR. Main areas of the research programme in the interests of the generation 4 reactors
436	J. Roglans-Ribas	USA	Versatile test reactor (VTR) project mission and status
357	A. Zagornov	Russian Federation	Multipurpose research facility MBIR and poly functional radiochemical complex (R&D complex) as a unique research platform
511	F. Heidet	USA	Versatile Test Reactor: conceptual core design overview

4.5.2. Session 5.2. Experimental Programmes I

Session 5.2 comprised seven presentations: two from India, one from Japan, one from Germany and three from Russian Federation.

A. Kumar (India) presented the outcomes of the experiments performed in the SILVERINA loop type facility regarding the sodium metal aerosols characterization in both high purity argon and nitrogen gas. The average charge acquired by sodium aerosols in argon and in nitrogen environment, respectively was determined theoretically by considering the mobility of argon and nitrogen in relation with the experientially measured sodium aerosols size and temperature of the cover gas region. It has been observed that the average charge on sodium aerosol is higher in the argon gas when compared to nitrogen, and it increases with sodium pool temperature, i.e., with the mean aerosol size. Also, the experiments shown that the deposition of aerosols in the narrow gaps, sodium pool surface, and Cs trapping for the aerosols could be enhanced under ionizing radiation. The aerosols migration from the liquid metals cooled reactors cover gas and their deposition is a topic that needs more investigations in order to better understand all the phenomena and mechanisms involved.

A. Mikheev (Russian Federation) presented the results of the thermal hydraulics tests supporting the design characteristics of the steam generator of the BREST-OD-300 reactor. Various models of the lead-heated steam generator have been considered. The tests carried out shown that under the nominal operating conditions, the design steam superheating is provided at the outlet of the steam-generating tube and, for all the operating modes (nominal, start-up, partial parameters) the steady-state operation of the steam generating channel is ensured up to a water flow rate of 4% of the nominal value. Moreover, the dependence of lead coolant heat transfer on the oxygen concentration in the lead has been confirmed experimentally. The results obtained as well as the author's recommendations are of main interest for lead fast reactor scientific community.

G.A. Sorokin (Russian Federation) presented the results of the calculated and experimental data comparison regarding the aspect of liquid metal boiling in fast reactor fuel assemblies which has a complex structure and is characterized by both stable and pulsation regimes with significant fluctuations of the parameters which could lead to a critical heat flux (CHF). Various liquid metal boiling experiments in simulated fuel assemblies (in a single fuel assembly and in a system of two fuel assemblies in a natural circulation loop) or in the pilot experiment devoted to a new technical solution (meaning a "sodium cavity" between the core and the upper axial blanket) have been performed. The in-house developed code SABENA-3D has been able to simulate the heat exchange process and the stability of the coolant circulation during liquid metal boiling both in a single FA and in a system of two parallel FAs in a natural circulation loop. All these investigations are needed to support the feasibility of a stable core cooling in the course of sodium boiling accident in the core, the associated limiting conditions as well as the generation of the experimental data needed for the validation of sodium boiling process model and the verification of the computational codes.

K. Aizawa (Japan) presented a study in support to the development of a robust DHRS able to ensure the long-term heat removal via natural circulation inside the reactor vessel. In this regard, water experiments using a scaled model (PHEASANT facility) based on the reactor vessel of an advanced loop-type SFR were conducted to investigate the thermal hydraulic phenomena in the reactor vessel. The natural circulation phenomena were investigated using the results of temperature and particle image velocimetry (PIV) measurements under the conditions of operating the dipped-type direct reactor heat exchanger (DHX) and reactor vessel auxiliary cooling system (RVACS), respectively. Also, the impact of temperature fluctuation on the PIV measurement has been quantitatively evaluated. Moreover, the measured temperature values confirmed the characteristics of the natural circulation flow field under RVACS operation.

N. Krauter (Germany) provided a detailed description of both the experiments performed aimed to investigate the possibility to use the Eddy Current Flow Meter (ECFM) as part of the safety instrumentation in order to monitor the coolant flow through subassemblies under normal operating conditions or to detect and locate blockages in case of a local freezing of the liquid metal. All the experiments have been carried out at room temperature using a liquid alloy of gallium, indium and tin. The measurements as well as the numerical simulations showed that the presence of the ECFM itself has a significant impact on the measured velocities, especially for flow angles above 40° with respect to the axis of the ECFM. The second type of experiments (using an array of ECFM sensors) investigated multiple cases with a different number and location of blocked subassemblies (SAs) and has demonstrated that these sensors can indeed be used to detect and localise blocked SAs as well as to monitor changes of the flow rate under normal operating conditions. Future experiments at higher temperatures and increased flow velocities of the liquid metal coolants will provide the needed knowledge and expertise concerning the qualification of ECFMs as part of the safety instrumentation of liquid metal cooled fast reactors.

Y. Sokolov (Russian Federation) presented the critical experiments carried out on FKBN-2 critical experiments facility (CEF) with ROMB cylindrical assembly specially designed for verification of Monte-Carlo calculation codes. The characteristics of the benchmark critical experiments carried out on the experimental facility are presented in detail. Moreover, for a better illustration of experimental capabilities of the FKBN-2 facility a brief description of previous and planned benchmark experiments aimed to verify Monte-Carlo calculation codes has been provided.

S.P. Pathak (India) presented the preliminary experimental investigations performed to ensure the controllability and operability of secondary circuit based decay heat removal system before implementation in a SFR reactor. The tests were performed by simulating the three operating phases of SSDHRS (Secondary Sodium based Decay Heat Removal System): (a) maintaining hot shutdown condition at various temperatures, (b) maintaining cold shutdown condition, (c) cool down phase following SCRAM replicating the OGDHR (Operation Grade Decay Heat Removal) deployment. The tests performed shown that both hot shutdown and cold shutdown conditions can be maintained by manually adjusting the air flow rate through the AHX (sodium-to-air heat exchanger). Thereby, it is possible to maintain of hot/cold shutdown conditions in the plant at any desired temperature by using a control logic which control the AHX outlet sodium temperature at a desired value by adjusting the air flow rate through AHX. Also, the tests showed that a variable frequency drive motor for AHX fans could be very useful for an efficient control of the heat removal performance of the system.

TABLE 21. PRESENTATIONS FROM SESSION 5.2 – EXPERIMENTAL PROGRAMMES I

Chairs: Daniela Gugiu and Yican Wu			
ID	Presenter	Designating Member State/Organization	Title of the Paper
466	A. Kumar	India	Estimation of mean charge on sodium metal aerosol in the argon and nitrogen gas environment during external gamma irradiation
148	A. Mikheev	Russian Federation	Thermohydraulic tests in justification of design characteristics of the BREST-OD-300 RP steam generator
187	G.A. Sorokin	Russian Federation	Experimental and computational studies of heat exchange for liquid metals boiling in fuel assembly models at accidental conditions
19	K. Aizawa	Japan	Investigation on natural circulation for decay heat removal in reactor vessel of sodium-cooled fast reactor
12	N. Krauter	Germany	Coolant flow monitoring with an Eddy Current Flow Meter at a mock-up of a liquid metal cooled fast reactor
379	Y. Sokolov	Russian Federation	Overview of critical experiments with fast metal cores held on assembly machine FKBN-2
476	S.P. Pathak	India	An experimental study on secondary sodium system based decay heat removal circuit of a sodium cooled fast reactor

4.5.3. Session 5.3. Experimental Programmes II

Session 5.3 comprised eight presentations: three from Italy, one from India, one from Romania one from Russian federation, one from France and one from USA.

G. Firpo (Italy) presented the status of ALFRED High priority R&D Needs. The aim of Gen IV nuclear system development is to excel in safety and reliability, having a very low likelihood and degree of reactor core damage and finally to eliminate the need for offsite emergency response. Advanced Lead Fast Reactor European Demonstrator (ALFRED) was addressed as mandatory in the European framework to foster the LFR commercial deployment, with the main goal to fill the gap from basic research to market penetration typically suffering from lack of investment.

M. Caramello (Italy) presented the results of the preliminary testing of ALFRED DHR System. Liquid metal cooled reactors face the challenge of coolant freezing that causes the reactor core to overheat. The authors presented a project called SIRIO to design, build and test a scaled experimental facility with respect to ALFRED's DHR system.

S.C.S.P.K. Krovvidi (India) presented the results of the in-sodium testing of AM350-welded disc bellows for FBTR control rod drive mechanism. Fast Breeder Test Reactor (FBTR), operational since

1985 is a 40 MW(th) loop type sodium cooled fast reactor. Material data required for design was generated by tensile testing of AM350-SCT1000 disc bellows at 530°C. Fatigue data was generated by 4-point correlation method. Bellows were manufactured in India. With 720 circumferential welds, around 300 m of weld, 0.2 mm depth of penetration and requirement of heat treatment, this work was first of its kind in India. Bellows were tested in air and sodium, and the design was qualified for deployment in reactors.

D. Gugiu (Romania) presented a study of implementation of LFR experimental infrastructures in Romania. The main actors of LFR agenda implementation are FALCON Consortium and the European support organizations which signed a MoU. The present as well as the envisaged financial resources needed for LFR technology development are represented by the National Fund for Research (PNCDI), European Cohesion and European Regional Development Funds (ERDF), HORIZON Europe, EURATOM as well as the Ministry of Energy which is financing annually the R&D Programme on "Advanced Nuclear Reactors and Fuel Cycles."

M. Grushko (Russian Federation) provided an overview of the experimental test facility to test a prototype of the air heat exchanger gate for the advanced BN reactor plant. Experimental test facility was designed to check the functionality of the prototype gate in the conditions similar to the standard ones and determine hydraulic resistance and air leaks in the closed position of the gate. All the objectives specified and those emerging in the course of the development and installation were achieved, which eventually allowed to create a high-temperature experimental test facility that meets all the necessary requirements and provides all the conditions required to test the gate prototype.

A. Quaini (France) presented the France-Japan collaboration on thermodynamic and kinetic studies of core material mixture in severe accidents of sodium cooled fast reactors. By 2019, experimental and calculation results as well as code analyses have been obtained in support of severe accident studies. Since 2020, task "Thermodynamic and Kinetic Studies of Core Material Mixture" focuses on improving models on material interactions at equilibrium as well as on kinetics of reactions to be used in severe accident simulation codes.

K. Weaver (USA) presented the Versatile Test Reactor (VTR) Experimental Capabilities. Normal Test Assembly (NTA) is a standard non-instrumented or passively instrumented open test assembly that is the same size as the drive fuel assembly. Dismountable Test Assembly (DTA) is a modified driver fuel assembly that will have an insert that replaces 7-19 pins of the driver fuel. Extended Length Test Assemblies (ELTA) are test assemblies that have a long stalk that extends through the reactor head, and typically have various instrumentation leads, for monitoring and controlling thermal hydraulic conditions. Rabbit Test Assembly (RTA) is a special test assembly for rapid transfer of capsules that contain experiment specimens.

M. Tarantino (Italy) introduced LFR Design and Technologies Development at ENEA. HELENA-2 is a loop type facility with working fluid under both forced and natural circulation regimes. It is conceived to test the ALFRED hottest fuel assembly thermal hydraulics, as well as flow induced vibration. Electrical Long-running Facility (ELF) is an electrical long-running and large-scale pool-type facility, designed to test the endurance and reliability under both forced circulation and natural circulation regimes, investigating in relevant scale the main thermal hydraulic phenomena occurring in a pool type configuration. Meltin'Pot is a research platform to study the fuel-coolant interaction, fuel dispersion and relocation in the coolant resulting from a severe accidental scenario, retention of fission products in lead and migration in cover gas. Hands-ON is a pool type experimental facility conceived to accommodate mock-up subassemblies to simulate the fuel handling operation.

TABLE 22. PRESENTATIONS FROM SESSION 5.3 – EXPERIMENTAL PROGRAMMES II

Chairs: Julia Kuzina and Christian Latgé

ID	Presenter	Designating Member State/Organization	Title of the Paper
345	G. Firpo	Italy	ALFRED High priority R&D Needs
324	M. Caramello	Italy	Preliminary testing of ALFRED DHR System
465	S.C.S.P.K. Krovvidi	India	Design, manufacturing and in-sodium testing of AM350-welded disc bellows for FBTR control rod drive mechanism
112	D. Gugiu	Romania	Implementation of LFR Experimental Infrastructures in Romania
197	M. Grushko	Russian Federation	Experimental test facility to test a prototype of the air heat exchanger gate for the advanced bn reactor plant. design and construction items
86	A. Quaini	France	France-Japan Collaboration on Thermodynamic and Kinetic Studies of Core Material Mixture in Severe Accidents of Sodium-Cooled Fast Reactors
423	K. Weaver	USA	Versatile Test Reactor (VTR) Experimental Capabilities
468	M. Tarantino	Italy	LFR Design and Technologies Development at ENEA: Status and Perspectives

4.5.4. Track 5: Poster Session

The poster session for Track 5 had five contributions. The presentations included topics such as Eddy Current Flow Meter flow rate measurements in liquid Sodium. The Eddy Current Flow Meter is an inductive flow rate sensor which can be used in many liquid metal applications and is well suited for operation in liquid metal cooled fast reactors. The evaluation of fluid-structure interaction effect as an important aspect of fast reactor structure safety evaluation as well as the decay heat removal system scaling verification and numerical pre-test analysis for ALFRED reactor were discussed. Further works focused on core system design and configuration of the Versatile Test Reactor and Fast Breeder Test Reactor were presented.

TABLE 23. POSTER PRESENTATIONS FROM TRACK 5

ID	Presenter	Designating Member State/Organization	Title of the Paper
13	G. Gerbeth	Germany	Eddy Current Flow Meter flow rate measurements in liquid Sodium at the SUPERFENNEC loop
244	D. Duan	China	Experimental investigation of the fluid-structure interaction effect between adjacent equipment supports in a fast reactor
312	R. Ganapathy	India	Conceptual Core configuration for increasing Power of Fast Breeder Reactor to 40 MWt
348	M. Caramello	Italy	ALFRED DHR system scaling verification and numerical pre-test analysis
366	A. Nelson	United States of America	Versatile Test Reactor: core system design requirements to support advanced reactor development

4.6. TRACK 6 – MODELLING, SIMULATIONS, AND DIGITIZATION

4.6.1. Session 6.1. Neutronics

Session 6.1 comprised eight presentations: one from Hungary, one from India, one from Mexico, one from USA, one from Italy, one from Switzerland, one from Russian Federation and one from France.

M. Szieberth (Hungary) presented the outcomes of comparison of calculation methods for lead cooled fast reactor reactivity effects. Sensitivity and uncertainty analyses were performed for the COMET critical assembly and the ALFRED demonstrator core with different codes and methodologies. Calculations for the COMET experiment confirmed validity of the applied tools for lead void coefficient calculation. Coolant temperature, cladding expansion and fuel temperature coefficients were investigated for ALFRED. Direct and perturbation theory calculations agree well, and sensitivity studies revealed the important effect of the Pb inelastic scattering. Monte Carlo results have high variance when small perturbations are calculated. Small geometry details are challenging in the homogenized deterministic models. Nuclear data uncertainties are significant for fast reactor applications.

A. Bachchan (India) presented the results of neutronic analysis of CEFR Start-up tests using FARCOB and ERANOS 2.1 code systems. In the first experiment about net criticality, experiment conditions are as follows: approach to the criticality by replacing mock-up assemblies step-by-step with actual fuel assembly. Reactor was brought to the critical state by loading 72 fuel assemblies. For super-critical extrapolation, the control rod RE2 was withdrawn step by step to three positions to reach super-criticality. As a result, both code systems underpredicted the core reactivity by only about 200 pcm. In the second experiment about control rod worth, the worth of each control rod was measured at operation core layout using 79 fuel assemblies in the core. Calculations show results in close agreement with the measured values.

R. Lopez-Solis (Mexico) presented the results of verification of the Simplified Spherical Harmonics (SPL) module of the neutron diffusion code AZNHEX through Neutronics Benchmark of CEFR Start-Up Tests. AZHNEX code is originally developed as a 3D neutron diffusion code for nuclear core analysis with hexagonal-z geometry. For validation of the solver, calculations were performed with regard to criticality, control rod measurements, sodium void, temperature reactivity, subassembly swap and integral coefficients. AZNHEX diffusion worked well with large cores but failed with smaller ones. Results were satisfactory for most cases under the current considerations of energy and spatial discretization.

F. Bostelmann (USA) presented a study of uncertainty analysis in modelling and safety analysis of SFRs using Neutronics as an OEAD/NEA benchmark. 18 contributions based on different methods and nuclear data libraries were submitted for the neutronics sub-exercises of the UAM SFR benchmark. Differences in uncertainties were mainly caused by differences in the covariance data; differences in methods and modelling played a minor role. Major differences were observed with Eddy Current Flow Meter (ECFM)/B-VII.0 compared to the other ENDF/B library releases. A report containing all detailed results is in progress.

D. Maurizio Castelluccio (Italy) presented the study of realisation of an adjusted nuclear data library based on Eddy Current Flow Meter (ECFM)/B-VIII.0 nuclear data evaluations for the ALFRED core. All the validation activities, inherently performed throughout the adjustment process, can be used to retrieve relevant information for the certification of the uncertainties affecting every calculated integral parameter of the ALFRED core, so strengthening the confidence in their estimates when verifying the compliance with the requirements imposed by the safety authority. Moreover, these results are also essential for leading the evaluation experts in updating the High Priority Request List (HPRL) of isotopes to draw the most appropriate strategy for planning new differential measurements, nuclear model refinements and data evaluation programmes.

J. Krepel (Switzerland) presented the experiments carried out on spatial interdependence of safety related effects in ESFR-SMART core. Coupled neutronics and Thermal Hydraulics (TH) simulations are important for conceptual studies as well as for safety assessment. Accuracy of the simulation tool should be selected according to the importance of the results. Standard solution uses multi-group XS as the coupling between TH and neutronics solver. The solver uses prepared XS to calculate the flux shape. Transients without sodium boiling, can be well addressed by TH and point kinetics. In transients with sodium boiling, which are CPU demanding from TH perspective, point kinetics is less precise.

I. Bukhtiiarov (Russian Federation) presented the solution of the nuclide kinetic equation for fast reactor in the OpenBPS code with options of choosing calculation method and uncertainties analyses. The OpenBPS code is designed as an open-source software product that is based on open-source tools and shared on GitHub. The programme provides the user with a set of options for preparing the initial task, manipulating the nuclear data and constants, selecting the method for solving, and filtering the result. The OpenBPS verification results indicate that the code can be applied at all stages of the fuel cycle, including operation in a fast neutron spectrum reactor.

L. Mesthiviers (France) presented the experiments focused on the actinide conversion capabilities of Molten Salt Reactors (MSR). Ternary salt allows for different core sizes and compositions. Chloride-based MSR has feature of good fission efficiencies for all plutonium isotopes. Intrinsic core safety is dominated by the density effect (very small Doppler effect). There are further studies planned on the use of a minor actinides and plutonium mix fuel.

TABLE 24. PRESENTATIONS FROM SESSION 6.1 – NEUTRONICS

Chairs: Emil Fridman and Xingkai Huo

ID	Presenter	Designating Member State/Organization	Title of the Paper
132	M. Szieberth	Hungary	Comparison of calculation methods for lead cooled fast reactor reactivity effects
316	A. Bachchan	India	Neutronics analysis of CEFR Start-up tests at IGCAR using FARCOB and ERANOS 2.1 Code Systems
52	R. Lopez-Solis	Mexico	Verification of the SPL module of the neutron diffusion code AZNHEX through Neutronics Benchmark of CEFR Start-Up Tests
272	F. Bostelmann	USA	Objectives and Status of Neutronics Sub-exercises of the OECD/NEA Benchmark for Uncertainty Analysis in Modelling for Design, Operation and Safety Analysis of SFRs
422	D. Maurizio Castelluccio	Italy	Realisation of an adjusted nuclear data library based on ENDF/B-VIII.0 nuclear data evaluations for the ALFRED core
36	J. Krepel	Switzerland	Spatial interdependence of safety related effects in ESFR-SMART core
166	I. Bukhtiiarov	Russian Federation	The solution of nuclide kinetic equation for fast reactor in the OpenBPS code with options of choosing calculation method and uncertainties analysis
147	L. Mesthiviers	France	Study on actinide conversion capabilities of Molten Salt Reactors (MSR)

4.6.2. Session 6.2. Thermal Hydraulics

Session 6.2 comprised of six presentations: two from Russian Federation, one from Slovakia, one from India, one from Italy and one from USA.

B. Kvizda (Slovakia) presented the main results of the thermal hydraulic benchmark exercise performed within the VINCO project and the results of the hot duct break for the 2 and 3 loop ALLEGRO model. The thermal hydraulic code-to-code benchmark exercise carried out during the EU VINCO project was focused on developing qualified models of ALLEGRO 75 MW demonstrator using various codes – RELAP5-3D, CATHARE2 and MELCOR. Two initiating events were selected for the benchmark exercise to cover both pressurized and depressurized transients: Loss of Coolant Accident with a 3-inch diameter and the total Station Blackout. Both steady-state and transient calculations have been compared and assessed. Based on the qualitative and quantitative assessment, the major distortions among the models, like the heat conductivity of the gas gap between the fuel pellet and the cladding, water-to-air heat transfer correlation and others, have been summarized, including their effect on the transient. The calculation results for hot duct break bypass transient showed that the three-loop ALLEGRO has better cooling capabilities than the two-loop ALLEGRO.

N. Pribaturin (Russian Federation) presented the results of experimental study of the oscillations of fuel rod simulators in the model of fuel assembly with spacer grids in the lead-bismuth eutectic coolant flow. Measurements were carried out on a 7 pins model. An annular channel with an equivalent diameter equal to the hydraulic diameter of the fuel assembly model was also used for measurements. In addition to the lead-bismuth coolant, a water coolant was also used, which made it possible to compare the results and determine the features of the vibrations of the fuel element simulator in liquids with different densities. It was shown that an increase in the fluid velocity leads to the appearance of additional noise in the region of low frequencies and to an increase of the maximum vibration amplitude. When the fluid velocity decreases, the vibration frequency of the rod approaches to the natural frequency of the rod in water.

D. Ganatra (India) presented a numerical model to simulate coolant boiling in a typical LMFBR subassembly. The model has been developed based on the formulation of SAS4A code. A multi-bubble slug ejection model was incorporated in the NaBOIL code to simulate liquid metal boiling due to the high liquid superheat and thermal conductivity of liquid metals. KNS L22 reference test was simulated to validate the code. NaBOIL results are compared with the results obtained by TRACE and SIMMER codes also. NaBOIL results agree very well with the predictions made by TRACE one-dimensional model. It is observed by comparison with TRACE and SIMMER codes as well as experimental results that these one-dimensional models were able to capture the phenomenological trend qualitatively but are inadequate to predict quantitatively and accurately the boiling behaviour of liquid metals accurately. However, it should be noticed that the Kompakter Natriumsiede-Kreislauf (KNS) bundle comprises of 37 pins only where flow through the wall sub-channels is much higher than that in actual reactor subassembly. With the increasing number of pins, the boiling behaviour of liquid metals can be more accurately simulated by one-dimensional models.

I. Piazza (Italy) presented the CFD model and calculations by ANSYS CFX v.19 code of the ALFRED FA in the nominal configuration and different degrees of internal blockages (30% and 50%). For the unblocked configuration results showed regular distributions with the velocity maximum and the temperature minimum at the centre of the subchannels and a strong local minimum in the side subchannels. Results for the lower spacer grid blocked showed the presence of the recirculating vortex extended downstream the blockage for both blockage levels (30% and 50%). This implies the presence of two temperature maxima, one at the beginning of the active region due to the vortex (local effect) and one at the end of the active region due to the lower mass flow rate in the blocked subchannels (global effect). A peak temperature of about 560°C and 680°C is foreseen at the end of the active region in the 30% and 50% blockage case respectively. For the 50% case, the maximum temperature overcomes the limiting temperature of 650°C fixed by the designers for the 15-15 Ti clad and therefore is not acceptable by design. The conclusion is made that the internal blockage in the active region dramatically affects the cladding temperature distribution behind the blockage. Therefore, the presence of a spacer grid in the active region should be avoided and the blockage is fixed in the first lower spacer grid.

O. Bovati (USA) presented the results of CFD calculations performed using the Reynold-Averaged-Navier-Stokes (RANS) approach with the $k - \omega$ Shear Stress Transport (SST) turbulence model to simulate an incompressible flow through a wire-wrapped fuel assembly using STAR-CCM+ code. The analysed results focused mainly on the axial and transverse pressure drops for a wide range of Reynolds numbers covering the transition and turbulent regimes. The results were compared with available experimental data provided by the Texas A&M experimental facility, and with the Upgraded Chen and Todreas Detailed (UCTD) correlation. It was found that this RANS model was able to predict the experimental and correlation friction factor values with a low relative error, since all the model predictions were within the experimental uncertainty interval. With respect to the transverse pressure drop, the bigger discrepancy occurred at low Reynolds number where the wire faced directly to the corner. All the remaining angular positions were in satisfactory agreement with the model predictions. It was concluded that the $k - \omega$ SST turbulence model can be used to predict pressure drop, both axial and transversal, for 61-pin fuel assemblies that use helical wire-wrapped as spacers. Also, it was concluded that considered RANS turbulence model should be used carefully to predict the transverse pressure drop at low Reynolds numbers because a higher relative error with respect to experiments was observed at these Reynolds numbers.

A. Palagin (Russian Federation) presented the new models of the system thermohydraulic code HYDRA-IBRAE/LM, developed in the "Codes of New Generation" subproject of "Proryv" project: the models for dispersed phase transport, post dryout heat exchange, lead coolant solidification, turbine model, water behaviour at supercritical pressure. Developed and implemented into the code multi-group model is based on kinetic equation for particles size distribution function, and the group method is used for its solution. Heterogeneous multi-group model allows describing processes, in which the bubble size distribution is important: downflow of heavy metal coolant where bubbles can be captured by the flow or move in the opposite direction depending on their size. The turbine model is based on universal thermodynamic relations connecting the values of enthalpy, entropy, density, mass fraction of water and pressure in turbine stages. In the model for the description of lead coolant solidification variation of solid lead mass is calculated in accordance with Stefan condition as being proportional to the difference of heat flux in the crust and heat flux in the liquid. The work was performed on providing the description of water behaviour at supercritical pressure by implementing the improved closure relations for wall friction and heat transfer.

TABLE 25. PRESENTATIONS FROM SESSION 6.2 – THERMAL HYDRAULICS

Chairs: Nastasia Mosunova and Jiri Krepel

ID	Presenter	Designating Member State/Organization	Title of the Paper
199	B. Kvizda	Slovakia	Recent thermal hydraulic studies of Gas Fast Reactor demonstrator ALLEGRO
107	N. Pribaturin	Russian Federation	Experimental modelling of a fuel element simulator vibration in a coolant flow
464	D. Ganatra	India	Development of coolant voiding model for fast reactor core
369	I. Piazza	Italy	ALFRED flow blockage analysis
370	O. Bovati	USA	CFD Simulations on a hexagonal 61-pin wire-wrapped fuel bundle with STARCCM+ and comparison with experimental data.
91	A. Palagin	Russian Federation	Progress in system thermohydraulic code HYDRA-IBRAE/LM models development for fast reactor simulation

4.6.3. Session 6.3. Multiscale and Multiphysics Calculations

Session 6.3 comprised of 7 presentations, two from Russia, one from the Netherlands, one from China, one from France, one from USA and one from Japan.

N. Mosunova (Russian Federation) presented the status of a set of 25 software products required for design, decision making and safety assessment of nuclear power plants with fast reactors, that has been developed in the Russian Federation within the subproject “Codes of New Generation” of the “Proryv” project. The developed software covers from neutron physics to the final stage of fuel cycle - SNF reprocessing and radioactive waste disposal. She presented the status and procedure for Verification and Validation, and mentioned that the experimental data from operating reactor facilities (BOR-60, BN-600, BN-800) and from studies of separate processes and phenomena, including the small-scale experiments performed in the Russian Federation were used for validation of the codes. In Q&A session, she explained that until now it is possible to share the codes neither for research nor for commercial activities, but it is possible to study this possibility in the future.

F. Roelofs (Netherlands) described the thermal hydraulic design and safety support activities of Nuclear Research and consultancy Group (NRG) with respect to liquid metal cooled reactors. The development of a tool allowing thermal hydraulic system analyses was presented. The SPECTRA code has been adapted to apply it to various liquid metals, a generic multi-scale modelling approach is under development coupling the SPECTRA code to CFD codes. The myMuscle: Multi-scale Thermal Hydraulics Coupling Tool was presented together with the current status of development and the future foreseen extensions. Finally, the fundamental activities on understanding and pragmatic engineering model development for turbulent heat flux were presented. In the discussion, Mr. Roelofs clarified that the tools are developed on C programming language in order to maintain the tools as generic as possible.

X. Jia (China) presented the validation work of SARAX code for the transient analysis of Sodium cooled Fast Reactors (SFRs) by means of the Experimental Breeder Reactor II (EBR-II) benchmark released by ANL. The SARAX code consists in a cross-section generation code named TULIP, a steady state neutronics calculation code named LAVENDER and a transient analysis code named DAISY. Some details about the point kinetics model used in the transient module were given in which local effects can be taken into account. SARAX gave comparable results with the benchmark and INL's results for steady state calculation. As for transient analysis, compared with experimental results and other institutes' results, SARAX presents good results. However, SARAX does not have the capability of system analysis at present, leading to the need to obtain input parameters for the transient calculation from the references. Questioned over the use of point kinetics, Ms. Jia said that it is due to the fact that point kinetics can produce results faster that 3D models.

B. Forno (France) presented a Multiphysics simulation based on a loosely coupled approach to take into account local Doppler feedback effect on power deformations, in the frame of a CRP to validate computational codes by means of a series of end-of-life tests performed in 2009 and 2010 in the PHENIX reactor before its definitive shutdown. The Serpent code together with the JEFF-3.1 library as used for the neutronic calculation and it was coupled to an in-house static thermal hydraulic solver. She showed that the coupling between neutron and thermal hydraulics solvers (with a regular exchange of the main coupled parameters such as fuel temperatures and neutron deposited powers for each axial node of each subassembly of the fissile core) were closer to the experimental values while comparing with uncoupled results (only neutronics calculation), although, it was presented just 2 neutronics-thermal hydraulics (NK-TH) iterations. The author was asked about the convergence of the NK-TH iterative process, and she argued that due to lack of time and computational resources no more iterations could be done, but initial calculations showed good improvement.

A. Moisseytsev (USA) presented the results of the application of the fast reactor safety analysis code SAS4A/SASSYS-1 to a selection of FFTF individual reactivity feedback steps, and compared the code predictions with the test data. In preparation for the passive safety demonstration tests in Cycle 8C, a series of individual reactivity feedback tests were performed in FFTF. The passive safety tests in Cycle 8A consisting of measurements of control rod positions at selected power and coolant conditions were presented and used to support validation of the reactivity feedback models in the SAS4A/SASSYS-1 safety analysis code that was developed at Argonne National Laboratory for transient simulation of liquid metal cooled fast reactors. The DEFORM-4 oxide fuel module was used to simulate the fuel pre-irradiation and transient behaviour, including fuel restructuring and growth, fission gas release, cladding deformation, and fuel-cladding gap size and thermal conduction. It was demonstrated that results for fuel Doppler and axial expansion reactivity feedbacks show good agreement with the tests. Some model refinements were identified and implemented in order to improve the agreement. The simple and detailed radial expansion models in SAS4A/SASSYS-1 were able to predict the results for Type 2 tests, but input adjustments were needed to represent uniform core radial expansion. Finally, the effects on core radial expansion due to temperature changes at both the grid plate and assembly load pads, as well as expansion of the control rod drivelines showed that cancelation of opposing feedbacks with similar magnitudes resulted in large relative discrepancies for net reactivity. Although the relative net reactivity discrepancies are larger than were observed for other test types, they are considerably smaller than the magnitudes of the positive control rod driveline expansion feedback and the negative feedback from radial core contraction that were observed. He pointed out in an answer that there are plans to evaluate other tests in the future.

V. Chudanov (Russian Federation) presented the development and implementation of one- and two-phase models to simulate heat and mass transfer processes in the separate elements of nuclear reactor. Those models are realized in the LES and DNS CONV-3D code in IBRAE RAN and that are part of the “Codes of New Generation” subproject of “Proryv” project. He presented a validation of the one-phase module against data of experimental data for various liquids, including lead and sodium used as coolants, in a wide range of Rayleigh numbers. For the two-phase model, the validation base includes experiments in which the heat and mass transfer and sodium boiling in the pipes were investigated. The two solvers were parallelized and can run in CPU and GPU as well. Several examples of code application for solving such problems as flow in fuel assemblies, tubes and ring channels, as well as natural convective flows in the elements of reactor were presented by the author. Also, results of two-phase flows modelling on the series of tests, including the problem of sodium boiling in a round pipe, were presented. In all cases a good agreement of numerical predictions with experimental data has been found.

N. Doda (Japan) presented the status and development of a multi-level simulation system at the Japan Atomic Energy Agency. The objective is to use detailed analysis codes for local phenomena of interest and coupling them with a plant dynamics analysis code in order to obtain evaluation results considering the mutual interaction without excessive conservativeness by successively updating the boundary conditions in coupling process. The coupled analysis method was presented using the plant dynamics analysis code Super-COPD and a subchannel analysis code named ASFRE, that has been developed to evaluate temperature distribution in a subassembly taking into account inter-subassembly heat transfer in radial direction of the core during the transient from forced circulation to natural circulation conditions. The numerical analysis on the EBR-II was analysed with two models in different level of detail for the specific subassemblies. The first model was the subchannel model of ASFRE in the coupling method and the second one was the channel model included in the whole core model of Super-COPD. Author demonstrated that the coupled analysis could predict transient temperature distribution in a subassembly in detail. The multi-level simulation for thermal hydraulics in a subassembly could be performed by changing the level of detail of the analysis model. Author expressed that for future work, it is planned to evaluate the sodium tests with PLANDTL test facility of JAEA as validation purpose of the coupled analysis method.

TABLE 26. PRESENTATIONS FROM SESSION 6.3 – MULTISCALE AND MULTIPHYSICS CALCULATIONS

Chairs: Armando Gómez Torres and Chirayu Batra (IAEA)			
ID	Presenter	Designating Member State/Organization	Title of the Paper
79	N. Mosunova	Russian Federation	Codes of new generation – sustainable platform for numerical modelling of installations in the Proryv project
62	F. Roelofs	Netherlands	Dutch Thermal Hydraulic Design and Safety Support for LMFRs
67	X. Jia	China	Verification of SARAX Code for the Transient Analysis of Sodium-cooled Fast Reactor
116	B. Forno	France	Phénix Control Rod Withdrawal test analysis using a multiphysics methodology
14	A. Moisseytsev	USA	Simulation of FFTF Individual Reactivity Feedback Tests with SAS4A/SASSYS-1 Code
84	V. Chudanov	Russian Federation	Current status of development of 3D DNS CONV-3D code: one- and two-phase flow models
24	N. Doda	Japan	Development of Multi-level Simulation System for Core Thermal-hydraulics Coupled with Plant Dynamics Analysis - Prediction of Transient Temperature Distribution in a Subassembly under Inter-subassembly Heat Transfer Effect

4.6.4. Session 6.4. Simulation Tools for Safety Analysis

Session 6.4 comprised of 8 presentations, one from Japan, one from China, one from Germany, one from France and four from the Russian Federation.

E. Usov (Russian Federation) presented Models of the integral EUCLID/V2 code for numerical simulation of severe accidents in a sodium cooled fast reactor with MOX and MNUP fuels. The code includes various modules for simulation of severe accident, thermal hydraulics, and three-dimensional neutron physics. The accident SAFR module with nitride fuel dissociation model were introduced. Further fission products release from melt pool modules and additional thermal hydraulics module HYDRA, based on two fluid modules to simulate two phase sodium flow were presented. BERKUT fuel rod module allows simulations of properties such as temperature distribution in fuel rod. Additional modules and models designed to simulate physical and chemical process characteristic of a sodium cooled reactor facility have been developed and implemented and validated in EUCLID/V2 Multiphysics code.

A. Uchibori (Japan) presented Development of Integrated Severe Accident Analysis Code, Severe-accident PhEnomenological Computational tool for TRansient Assessment (SPECTRA) for Sodium cooled Fast Reactor (SFR). SPECTRA has been developed for an integrated analysis of in-vessel and ex-vessel phenomena during severe accidents in SFRs. The in-vessel and ex-vessel modules are coupled by exchanging the amount of leaked sodium and debris at every time step. SPECTRA was successfully applied to the Loss Of Reactor Level (LORL) event in which in-vessel and ex-vessel phenomena progress concurrently. The future work includes extension of model applicability, for example, coupling of CFD and lumped mass model for in-vessel coolant behaviour and application to dynamic Probabilistic Risk Assessment (PRA).

A. Sorokin (Russian Federation) presented a brief review of AEROSOL/LM module, designed to simulate the behaviour of FPs in the circuits and rooms of NPP with FRs with sodium and lead coolants. The results of test calculations showed the possibility of modelling the behaviour of multi-components aerosols of FPs, taking into account changes in the density of the aerosol material depending on the composition. The results of the validation of the AEROSOL/LM module showed the possibility of modelling the processes of sodium combustion in a jet and pool with the subsequent formation of aerosols of sodium combustion products in NPP rooms, including their interaction with FP aerosols and deposition on the surfaces of walls and floors. The AEROSOL/LM module as part of the EUCLID/V2 code was developed as part of the New Generation Codes Project “PRORYV”.

Y. Liu (China) presented the result of analysis of the natural circulation capacity of decay heat removal system in pool-type sodium cooled fast reactor. Placing DHX in the cold pool promotes natural circulation within the assemblies. Meanwhile, placing DHX in the hot pool facilitates natural circulation in inter-wrapper region. Analysis whether the location of DHX in the hot pool affects the decay heat removal by natural circulation and analysis the cooling capacity of DHX in the cold pool under Core Disrupt Accident (CDA) condition are anticipated as future works.

W. Klein-Hessling (Germany) presented the state of Regulatory Perspectives on Analytical Codes and Methods for Advanced Reactors. Outcomes of the OECD/NEA working group on the safety of advanced reactors were presented, based on a multi-national survey of best practices and methods being used in Canada, France, Germany, Italy, Russia, UK and USA. Many common approaches of confirmatory analysis were identified such as needs related to verification and validation, estimation of uncertainties and conduction of confirmatory analyses. There was, however, not a complete agreement on how to identify and address gaps in knowledge basis amongst the countries. Notable differences between countries were regarding certification of codes by regulatory authority and expectation on user qualification. The development and qualification of codes and methods should be regarded thoroughly in Gen-IV reactor development projects.

D. Veprev (Russian Federation) presented the development of models of the integral EUCLID/V2 code for numerical modelling of different regimes of lead cooled fast reactor. Additional modules and models designed to simulate physical and chemical processes characteristic of a lead cooled reactor facility have been developed and implemented in the EUCLID/V2 multi-physics code. New models of solid phase impurity transport, steam generator tube rupture, fission product source, nitride fuel dissociation, lead melt and concrete interaction, and the lead freezing model were all implemented. Verification and validation of the new models is ongoing, and these models will make it possible to simulate the behaviour of a reactor facility with a lead coolant both in normal operation conditions and anticipated operational incidents.

E. Ivanov (France) presented the Target Accuracy Requirements and an evidence-based background for molten salt fast reactor (MSFR) safety assessment. The presentation was aimed at how to best identify and address gaps in the experimental data validation basis for MSFRs given the limited set of representative experiments with this system. The importance of translating experimental results into benchmarks was highlighted, as well as identifying the limits of what conclusions could be drawn from the benchmarks. The added value of a particular set of information was discussed in the context of quantifying knowledge gaps.

R. Chalyy (Russian Federation) presented the development status of SOCRAT-BN integral code: Development, validation, and current status. Improvements to the SOCRAT-BN code were highlighted, the development of SOCRAT-BN code has been fully completed at present. The code has been certified by Rostekhnadzor and is currently used to justify the safety of sodium cooled fast breeder reactor plants during off normal operation, design basis and beyond design basis accidents. The code is anticipated to be able to simulate an accident from the initial event to the formation of the source of fission products. IBRAE RAN specialist is providing technical support to user in terms of improving code models, diagnostic systems, and user interfaces.

TABLE 27. PRESENTATIONS FROM SESSION 6.4 – SIMULATION TOOLS FOR SAFETY ANALYSIS

Chairs: Evgeny Ivanov and Ian Hill

ID	Presenter	Designating Member State/Organization	Title of the Paper
85	E. Usov	Russian Federation	Models of the integral EUCLID/V2 code for numerical simulation of severe accidents in a sodium-cooled fast reactor with MOX and MNUP fuels
21	A. Uchibori	Japan	Development of Integrated Severe Accident Analysis Code, SPECTRA for Sodium-cooled Fast Reactor
95	A. Sorokin	Russian Federation	Aerosol module for modelling of the fission product behaviour in FR cooling circuits and NPP compartments
415	Y. Liu	China	Analysis of the natural circulation capacity of decay heat removal system in pool-type sodium-cooled fast reactor
350	W. Klein-Hessling	Germany	Regulatory Perspectives on Analytical Codes and Methods for Advanced Reactors
97	D. Veprev	Russian Federation	Models of the integral EUCLID/V2 code for numerical modelling of different regimes of lead-cooled fast reactor
440	E. Ivanov	France	Target Accuracy Requirements and an evidence-based background for MSFR safety assessment
93	R. Chalyy	Russian Federation	SOCRAT-BN integral code: Development, validation and current status

4.6.5. Session 6.5. Integrated Analysis and Digitalization

Session 6.5 comprised of 7 presentations, one from Switzerland, one from USA, five from the Russian Federation.

A. Fedorovskii (Russian Federation) presented the digital technologies developments in Proryv project. The project is implemented by the State Atomic Energy Corporation ROSATOM and designed to solve a complex of the most complex innovative tasks, the implementation of which required the use of advanced information technologies and solutions. Digital twins of objects are created for consolidation, timely verification and effective use in the future from the earliest stages of development. With the help of digital twins, continuous analysis and optimization of all key indicators of the designed and constructed facilities of the Proryv project is carried out. Thanks to the use of digital twins, it has already been possible to justify and work out a large number of measures to optimize the technical and economic indicators of the project.

A. Jiménez Carrascosa (Switzerland) presented the development of an artificial neural network (ANN) for predicting spatial interdependencies of reactivity effects in sodium cooled fast reactors. In this work, an ANN is developed for predicting sodium void effects in a large sodium fast reactor core and their spatial interrelations. The ultimate goal is to provide more realistic inputs to the thermal hydraulics code TRACE for point-kinetics-based transient analysis of the most recent ESFR core concept. With that goal, an ANN is developed and trained to provide the global sodium density effect, receiving as input the normalized sodium density at the different regions of the core. Local reactivity effects are computed using ERANOS deterministic code for an extensive set of combined scenarios in order to train the ANN. The developed model can predict the reactivity evolution taking into account the spatial interrelations between active core and sodium plenum.

A. Ushatikov (Russian Federation) presented the application of digital twin of fast reactor plants for control system algorithm testing. The developed technology of creating digital twins of fast neutron reactor plants allowed increasing the rate of development and validating the safety of newly developed designs, reducing the cost of expensive testing and debugging at the stage of start-up and adjustment, and cutting down the design time. Currently, JSC "Afrikantov OKBM" extensively uses the fast reactor plant digital twin in a number of R&D activities finding application to meet a number of objectives. Implementation of this technology at early design stages showed its efficiency and allowed cutting down the time of reactor plant development.

A.V. Moiseev (Russian Federation) presented the concept of the BREST reactor with lead coolant and dense heat-conductive nitride fuel that envisages the development of an equilibrium core with complete breeding of fissionable nuclides in the core without a blanket compensating for reactivity reduction due to fuel burnup and fission-product buildup. To carry out computational studies of the BREST reactor core, a design code system is used, which includes the FACTBR diffusion software system, the MCU-BR software tool based on the Monte-Carlo method, and the IVIS-BR thermophysical module. The developed and validated design code system and the results obtained in computational studies may be used to support design solutions for the BREST-OD-300 reactor and commercial lead cooled fast reactors.

D. Wise (USA) presented the work of Westinghouse to develop its Next Generation high-capacity nuclear power plant based on lead cooled fast reactor technology. By leveraging its long experience in nuclear power plant commercialization as well as strategic domestic and international partnerships established to complement capabilities most effectively, Westinghouse is progressing the plant's design and its business and delivery model. With a power output of approximately 950 MW(th) (~460 MW(e)), the Westinghouse LFR is a competitive, medium-size, simple, scalable and passively safe plant harnessing a liquid lead cooled, fast neutron spectrum core operating at high temperatures in a pool configuration reactor encompassed by a passive heat removal system (PHRS). The PHRS design encompasses the LFR vessel and relies on a pool of water surrounding the guard vessel to effectively remove decay heat during postulated accidents when the normal decay heat removal system is not available. The calculation results were presented using the thermal hydraulics code GOTHIC and matched quite well with the analytical results. As next steps, it is expected to perform some CFD calculations to compare and validate the results.

S. Belov (Russian Federation) presented two types of mixed uranium-plutonium fuels that are considered for BN-1200 reactor within the framework of "Proryv" project: MOX-fuel and advanced mixed uranium-plutonium nitride (MNUP) fuel. The reactivity margin of the start-up core shall be predicted considering its possible deviation caused by fuel manufacturing tolerance and by uncertainties of core critical parameters estimation. For this reason, in the start-up core design, even increased fuel enrichment may be considered, and, in any case, measures to compensate possible excess reactivity margin due to development of an appropriate core arrangement shall be provided. Possible methods were proposed to compensate excess reactivity margin of the start-up core. Basic approaches to form the start-up core and the core arrangement in the case of MOX-fuel and MNUP fuel application were presented.

S. Belov (Russian Federation) presented the proper core design to solve problems of the BN-800 transition from a hybrid core consisting of fuel subassemblies (FSAs) with pellet-type uranium oxide fuel and FSAs with pellet-type and vibro-packed MOX fuel to a core completely loaded with pellet-type MOX fuel (full MOX-fuel core). The transition period is characterized by step-wise increase of neutron flux density in the core caused by nuclear distinctive features of plutonium relative to uranium-235. Change of ratio of quantities of FSAs of different types with different hydraulic resistance leads to correspondent re-distribution of sodium flowrates through them and to change of the total hydraulic resistance of the core. The data on sodium pressure drop over the core and on the temperature state of FSAs during transition period were presented. The FSAs operation parameters do not exceed the justified values. No reactor power limitation is required during reactor operation in the transition period.

TABLE 28. PRESENTATIONS FROM SESSION 6.5 – INTEGRATED ANALYSIS AND DIGITILIZATION

Chairs: Youqi Zheng and Chirayu Batra (IAEA)			
ID	Presenter	Designating Member State/Organization	Title of the Paper
88	A. Fedorovskii	Russian Federation	Digital technologies for project development ODEC and PEC and digital twins
46	A. Jiménez Carrascosa	Switzerland	Development of an Artificial Neural Network for predicting spatial interdependencies of reactivity effects in Sodium Fast Reactors
129	A. Ushatikov	Russian Federation	Application of digital twin of fast reactor plant for control system algorithm testing
364	A.V. Moiseev	Russian Federation	Computational Studies of Advantages of Lead-Cooled Fast Reactor Core
521	D. Wise	USA	Passive Heat Removal System Analysis for the Westinghouse Lead Fast Reactor
208	S. Belov	Russian Federation	Approaches to form the BN 1200 core start loading using MOX-fuel and MNUP-fuel
209	S. Belov	Russian Federation	Distinctive features of the BN-800 core in the course of transition to complete MOX-fuel loading

4.6.6. Session 6.6. Fuel Performance and Material Modelling

Session 6.6 comprised of seven presentations, two from Switzerland, one from France, one from China, one from USA, one from Russian Federation and one from India.

C. Fiorina (Switzerland) presented the implementation of the SCIANTIX inert gas behaviour model into the OFFBEAT multi-dimensional fuel performance code, including its preliminary validation against experimental fuel rods from the IFA-432 assembly, Super-Ramp programme and Risø-3 experiment. Obtained results can be considered as satisfactory and in-line with other results obtained by various authors.

J.B. Genin (France) presented the results of the evaluation of the full set of parameters for the corrosion product contamination code OSCAR-NA. The OSCAR-Na code has been developed during the last decade to calculate the mass transfer of corrosion products and related contamination in the primary circuit of sodium fast reactors (SFR). The full set of parameters for the corrosion product has been evaluated for each element (Fe, Ni, Cr, Mn, Co) as a function of temperature through comparison of simulations with measurements in sodium loops and in sodium fast reactors. A satisfying validation of the OSCAR-Na code is obtained for the operational domain of a sodium fast reactor.

J. Dietz (Switzerland) presented the of the MSR fuel cycle and thermo-dynamics simulation (for the initial salt composition and its development as the reactor operates), with a combination of two main tools - the EQL0D and GEMS codes. For GEMS, the HERACLES database is validated and extended, and equilibrium salt compositions are analysed. In EQL0D, simplified cases are created to handle the combinations of different fertile and carrier salt options. Further additions to the HERACLES database will extend the capability of GEMS, while more intricate EQL0D cases will provide elemental compositions that are truer to real designs.

Y. Liu (China) presented the results of the study of the fluid-structure interaction of narrow gaps between thin-wall coaxial structures (main pumps and the main vessel) with their thermal shield in fast reactors. The fluid pressure and acceleration distribution of such structures under different modal shapes are measured. A data processing method is established to transfer experimental results to the added mass. The correlation between added mass and circumferential wave number is obtained, which can be useful in the structural assessment of key equipment with a fluid-structure interaction effect. The results should facilitate the seismic analysis of the FR design.

M. C. Messner (USA) presented a complete statistical design and analysis method for primary load, steady state creep, suitable for the design of high temperature fast reactor components. The goal of the method is to provide a geometric design that meets a target reliability, assuming creep rupture under steady load as the dominant failure mechanism. The statistical creep rate and rupture distributions provide the underlying data for the design analysis. The complete method applies the Monte Carlo approach to sample the creep rate and rupture stress distributions, providing a probabilistic assessment of the life of the component. The statistical analysis can be compared to a deterministic design analysis to quantify the design margin in terms of the component reliability.

A.V. Zadorozhnyi (Russian Federation) presented the simulation results of the thermomechanical behaviour of fuel rods with MNUP fuel for Russian fast reactors with a liquid metal coolant. The simulations carried out using the engineering and advanced versions of the fuel performance code BERKUT, and comparison of calculated results with the results of post-irradiation examination (PIE). The results on the fission products release obtained by the advanced version are in very good agreement with the PIE results, for the results of the engineering version the agreement is worse, and the values obtained for the maximum value of the initial fuel-cladding gap noticeably exceed the PIE results. The radius-averaged volumetric fuel swelling calculated by the advanced version is in good agreement with the results predicted by the parametric dependence for the times when most of the major transients in the fuel stopped.

R.K. Maity (India) presented the results of the computational fluid dynamics study for estimation of failed fuel location, detected by a global delayed neutron detection system in Failed Fuel Location Module, in the sodium cooled fast reactor. The results demonstrated the reliable and accurate sample collection for failed fuel localization.

TABLE 29. PRESENTATIONS FROM SESSION 6.6 – FUEL PERFORMANCE AND MATERIAL MODELLING

Chairs: Carlo Fiorina and Anzhelika Khaperskaia (IAEA)

ID	Presenter	Designating Member State/Organization	Title of the Paper
51	C. Fiorina	Switzerland	Simulation of fission gas release in the 3-D fuel performance code OFFBEAT
83	J.B. Genin	France	First fully adjusted set of parameters for the corrosion product contamination code OSCAR-Na
34	J. Dietz	Switzerland	MSR Fuel Cycle and Thermo-Dynamics Simulations
151	Y. Liu	China	The fluid structure interaction of narrow gaps between thin-wall coaxial structures in fast reactors
50	M. C. Messner	USA	A statistical design method for steady state creep applied to Grade 91 components
94	A.V. Zadorozhnyi	Russian Federation	Mechanistic code BERKUT-U: self-consistent modelling of fuel rods thermomechanical behaviour and processes in the fuel of fast breeder reactors
459	R.K. Maity	India	Computational fluid dynamics study for estimation of dilution for failed fuel location system

4.6.7. Track 6: Poster Session

The poster session in Track 6, the largest Track had 14 contributions. The topics covered a broad range of development and issues involving neutronics, thermal hydraulics and others, including neutronic calculation of the China Experiment Fast Reactor core, modelling of the coolant region in the ALFRED core in case of thermal expansion, sensitivity and uncertainty analysis of the key safety relevant reactivity coefficients for the ALFRED core, characterization of the Molten Chloride Fast Reactor fuel cycle options and reprocessing of the spent nuclear fuel in the chloride melt, radiation safety assessment of reactor and fuel cycle facilities, material activation calculation, application of model based system engineering in design of Digital Fast Reactor nuclear power plant, thermal hydraulic characteristics of the reactor plant based on the operation experience of the BN-800 reactor and Accelerator Driven Systems for Energy Production and Uranium-233 breeding.

TABLE 30. POSTER PRESENTATIONS FROM TRACK 6

ID	Presenter	Designating Member State/Organization	Title of the Paper
33	J. Krepel	Switzerland	Characterization of the Molten Chloride Fast Reactor fuel cycle options
53	R. Lopez-Solis	Mexico	Verification and validation of the CEFR Serpent model for the generation of reference solutions and Cross Sections database for the deterministic code AZNHEX
96	A. Belov	Russian Federation	Calculation of the materials activation with BPSD code
98	V. Bereznev	Russian Federation	Integral code COMPLEX for radiation safety assessment of reactor and nuclear fuel cycle facilities
110	A. Leshchenko	Russian Federation	Experience of Using CFD Models for Development of High-Temperature Furnace Equipment for Fabrication of Mixed Nitride Uranium-Plutonium Fuel Pellets
135	A. Moise	Romania	Neutronic Calculation of CEFR Core using Different Nuclear Data Libraries
165	A. Ali	Egypt	ADS for Energy Production and 233U breeding in HEU-Thorium Oxide system
179	I. Fadeev	Russian Federation	Elaboration of the thermal-hydraulic characteristics of the reactor plant based on the operation experience of the power unit with BN-800 reactor
216	S. Rogozhkin	Russian Federation	Development of the technical approach for research of the sodium coolant current in the integral type reactor
308	G. Yujie	China	Application of Model Based System Engineering in Design of Digital Fast Reactor Nuclear Power Plant
416	R. Pergreffi	Italy	Modelling of the coolant region in the ALFRED core in case of thermal expansion
421	D.M. Castelluccio	Italy	ENDF/B-VIII.0 nuclear data sensitivity and uncertainty (S/U) analysis of key safety-relevant reactivity coefficients for the ALFRED core

TABLE 30. POSTER PRESENTATIONS FROM TRACK 6 (CONT.)

ID	Presenter	Designating Member State/Organization	Title of the Paper
460	R. K. Maity	India	Integrated thermal hydraulic analysis of Hot and Cold Pools of a liquid sodium cooled 600 MWe fast reactor
542	A. Potapov	Russian Federation	Thermodynamic simulation of the oxidation processes at the reprocessing of spent nuclear fuel in the LiCl-KCl melt

4.7. TRACK 7 – SUSTAINABILITY: ECONOMICS, ENVIRONMENT AND PROLIFERATION

4.7.1. Session 7.1. Sustainability: Economics, Environment, and Proliferation Concerns

Session 7.1 comprised eleven presentations: one from China, one from France, one from IAEA, eight from Russian Federation.

A. Egorov (Russian Federation) presented a model of global nuclear energy system deployment with fast and thermal reactors in a partially closed fuel cycle. The goal of this study is to find the global nuclear energy system structure that is optimal from the point of view of economic criteria. The results of the study show that in the conditions of lower capital costs of thermal neutron reactors compared to fast reactors, low prices for natural uranium and low cost of spent nuclear fuel (SNF) storage, the development of a global nuclear energy system (NES) based on a once-through nuclear fuel cycle (NFC) with thermal reactors and uranium fuel will dominate until the 70s of the 21st century. However, increase in the price of natural uranium associated with the system growth and increase in the costs of fuel management at the final stage of the once-through NFC, leads to an economically justified commissioning of fast reactors and partially closed NFC. The paper identifies the factors that most significantly affect the structure of the global system optimized for economic criteria.

D. Tolstoukhov (Russian Federation) presented the key aspects of competitiveness for industrial energy complex with Fast Reactors and closed nuclear fuel cycle. The competitiveness of nuclear generation is determined not only by technical and economic indicators of promising NPP projects and corresponding NFC facilities. The competitiveness of NPPs is also influenced by technical and economic indicators of alternative energy technologies, taking into account the prospects for their optimization (improvement), the dynamics of prices for hydrocarbon fuel and the macroeconomic environment (discount rate). Current competitiveness requirements for NPPs with fast reactors and closed NFC, achievement of which ensures sustainable economic efficiency in comparison with alternative energy technologies, were reviewed.

D. Settimo (France) presented a study conducted to define a functional description and a sketch of the commercial industrial French Sodium Fast Reactor 1000 MW(e) (SFR 1000). In a second part, the author presented an economical comparison of a commercial SFR in France and an equivalent PWR, based on the evaluation of investment cost of the SFR 1000 and of a commercial European Pressurized Reactor (EPR). Finally, some technical proposals to reduce the cost of the commercial SFR, given with the needs of R&D or design assessment to be performed were presented.

A. Chebeskov (Russian Federation) presented the specific features of the export of Russian technologies of fast reactors and a closed nuclear fuel cycle. In the presentation, the author analysed the potential export of nuclear power plants with fast reactors, including nuclear fuel, as well as closed nuclear fuel cycle facilities, taking into account both the technical and economic indicators of such power units, and the existing system of International supply regime in the field of nuclear energy technologies. The paper discussed the main features of fast reactor technology in comparison with the existing technology of exported thermal reactors in the context of the implementation of the International supply regime in the nuclear power field.

P. Li (China) presented the development status of commercial SMRs and its experience to China. The paper summarized the technical route, government support, investor interest, investigated the present situation and the development trend of small nuclear power reactors in Russia and the United States. Therefore, considering the comprehensive evaluation of the safety and economy of all kinds of commercial SMR, the author concluded that developing a small modular lead-bismuth cooled fast reactor is a better choice in China.

I. Zhuravlev (Russian Federation) presented the effect of reactor technology on economics of SMR projects. Using regression analysis of the LCOE values published by vendors of SMR projects and own calculations, the author identified the effects of the scale, "learning" and reactor technology on the economics of SMR. The competitiveness of SMRs based on advanced reactor technologies beyond 2030 depends on the future severity of nonproliferation issues for potential customers. Depending on this, either HTGRs with a thermal spectrum (the most proliferation-resistant due to the use of TRISO fuel) or reactors with a fast neutron spectrum can receive priority.

A. Egorov (Russian Federation) presented a comparative multi-criteria analysis of scenarios of the Russian nuclear energy development in the context of uncertain knowledge about the future. The paper showed an approach to the calculation-based justification of the phased transition of nuclear power industry in Russia to the regime of a two-component nuclear energy system with a centralized closed nuclear fuel cycle, based on the use of a multi-criteria analysis method.

E. Marova (Russian Federation) presented the efficient fuel supply of a two-component nuclear system with VVER-type and BN-type reactors. The paper considered various scenarios of the nuclear industry development and provides a comprehensive comparative analysis of their efficiency. Multicriteria comparative analysis showed that the rating of the two-component scenario with closed nuclear fuel cycle is significantly higher than the single-component scenario with open nuclear fuel cycle.

N. Salnikova (Russian Federation) presented the export potential and commercialization conditions of fast reactors considering non-proliferation items. The FR export potential is limited because of the possibility to use the FR with breeding ratio more than 1 to produce weapon-grade plutonium even if a country-recipient has no power unit with the FR for spent nuclear fuel reprocessing at the moment of delivery. The paper considered possible export scenarios for FR-based power units considering nuclear weapon non-proliferation requirements and obligatory conditions of this export implementation. Considering the results of analysis of limitations, alternative scenarios of FR export potential realization providing full use of distinctive features of the FR technology while meeting the non-proliferation requirements were proposed.

O. Komlev (Russian Federation) presented the technological support of the non-proliferation for SVBR-100 fuel cycles. To solve the problem of non-proliferation of nuclear fissile materials, both institutional measures and technological support measures are used. Reactors implementing different fuel cycles, in order to ensure IAEA safeguards, require significant control during their deployment in non-nuclear countries due to the presence of significant quantities of plutonium, americium and neptunium in the fuel of such reactors. To increase the technological support for non-proliferation of nuclear materials, various open NFCs and closed NFCs for SVBR-100 were considered. These fuel cycles deal with nuclear fuel with significant values of heat generation in plutonium and americium as well as nuclear fuel without significant quantities of neptunium in SNF.

A. Bychkov (IAEA) presented the INPRO project studies on the double-component nuclear power systems with the closed fuel cycle and fast reactors. The double-component system (LWR+FR+CFC) was one concept for sustainable production of energy in INPRO initial phases. The author presented the last published INPRO reports included consideration of Multicomponent NP Systems, and the step forward INPRO pilot studies.

TABLE 31. PRESENTATIONS FROM SESSION 7.1 – SUSTAINABILITY: ECONOMICS, ENVIRONMENT, AND PROLIFERATION

Chairs: Brian Boyer (IAEA) and David Settimo

ID	Presenter	Designating Member State/Organization	Title of the Paper
27	A. Egorov	Russian Federation	Modelling the optimal economic structure of a global deploying nuclear power system with fast and thermal reactors in a partially closed nuclear fuel cycle
318	D. Tolstoukhov	Russian Federation	Key aspects of competitiveness for industrial energy complex with FR and closed NFC
128	D. Settimo	France	Technical and economical features of commercial sodium fast reactor in France
20	A. Chebeskov	Russian Federation	Specific features of the export of Russian technologies of fast reactors and a closed nuclear fuel cycle
220	P. Li	China	Development status of commercial SMRs and its experience to China
142	I. Zhuravlev	Russian Federation	Effect of Reactor Technology on Economics of SMR Projects
26	A. Yegorov	Russian Federation	Comparative multi-criteria analysis of scenarios of the Russian nuclear energy development in the context of uncertainty knowledge about the future
146	E. Marova	Russian Federation	Effective fuel supply of two-component nuclear energy system with VVER-BN reactors
71	N. Salnikova	Russian Federation	Export Potential and Commercialization Conditions of Fast Reactors Considering Non-Proliferation Items
385	O. Komlev	Russian Federation	Technological support of the non-proliferation for SVBR-100 fuel cycles
550	A. Bychkov	IAEA	The INPRO project studies on the double-component nuclear power systems with the closed fuel cycle and fast reactors: past and future

4.7.2. Track 7: Poster Session

The small track 7 poster session involved four presentations focused on sustainability, involving results of a case study on a comparative evaluation, and ranking of six power generation technologies including two nuclear, two fossil fuel and two renewable power generation options and work on the current concerns of the hydrogen community in finding solutions based on scientific fundamentals for hydrogen production using nuclear energy; economic aspects such as approaches used to estimate the commercial advantages of commercial fast reactor application both within energetic system and in the international context and at last proliferation. The work on proliferation presents the BREST pilot demonstration energy complex being built in Russia to show the potential reduction in nuclear proliferation risks through the incorporation of technical features.

TABLE 32. POSTER PRESENTATIONS FROM TRACK 7

ID	Presenter	Designating Member State/Organization	Title of the Paper
133	M. Roslaya	Russian Federation	Two-Component Energy Industry under Conditions of Closed Nuclear Fuel Cycle: Economic Benefits
140	S. Kviatkovskii	Russian Federation	Sustainability of nuclear and non-nuclear power generation options under Russian conditions: a comparative evaluation study
381	I. Iordache	Romania	Nuclear Hydrogen and Fast Reactors
548	V. Kuchinov	Russian Federation	Export of RBN with SNCD and nuclear proliferation risks

4.8. TRACK 8 – COMMISSIONING, OPERATION, AND DECOMMISSIONING

4.8.1. Session 8.1 SFR Commissioning, Operation, and Decommissioning

Session 8.1 comprised of eleven presentations, one from France, seven from India and three from Russian Federation, on fast reactor commissioning, operation, decommissioning and development of various inspection tools for SG.

J. Winston (India) presented the development of Reactor Core Viewing System (RCVS) for the Pre-Commissioning Stage Inspection of Reactor Core Components of PFBR. This presentation described the testing and qualifying for deployment in reactor at room temperature. The RCVS would fulfill its role for a pre-service inspection before sodium filling in the main vessel.

A. Kumar (India) presented the experience in preheating of PFBR reactor assembly. This presentation described the preheating strategy adopted for PFBR, which is increasing the temperature at a maximum rate of 5°C/day to achieve uniform temperature rise among internals. Based on the preheating carried out in PFBR, the total activity could be split into two stages. The first stage involves preheating of IHX to 150°C for filling in the secondary sodium main circuit. The second stage involves preheating of entire reactor assembly to 150°C for filling sodium into the main vessel.

N. Sahu (India) presented the commissioning and operating experience for secondary sodium systems and auxiliaries of PFBR. This presentation described the demonstration of the safe, continuous, and reliable operation of the Electromagnetic pumps and Secondary Sodium Pumps. Commissioning and operation brought out the need for few design changes not only to solve completely the problems faced but also to improve the system performance and safety. The experience has proved that the functional requirements of Secondary Sodium systems are fulfilled in all the expected configurations.

J. Jose (India) presented the development of Advanced In-situ Calibration and Probe Release Mechanism for PFBR SG Inspection System (PSGIS). This presentation described in detail the design, operation, and qualification of the developed devise for PSGIS. The Advanced In-situ Calibration has been designed and incorporated into the existing inspection system by attaching the calibration tube in the Device Deployment Module (DDM) along the pathway of the probe. Using Probe Release Mechanism, the probe can be inserted and retracted without decoupling the PSGIS device from the SG module. This feature helps in reducing the downtime and eases the process of probe change and calibration.

D. Villani (France) presented the overview of treatment of sodium of Superphenix fast breeder reactor. This presentation described that the dismantling of primary vessel of SPX are going to achieve successfully, following the treatment of the bulk sodium and the residual sodium. And the treatment of sodium (in mass and residual) allowed a safer dismantling and an evacuation of the waste without problem. The risks incurred during dismantling are greatly reduced by eliminating the anoxia risk, the sodium risk, and the chemical risk (soda). The tritium risk has been completely managed by allowing its elimination by controlled release to the atmosphere at the stac.

M. Thangamani (India) presented the overview of operating experience of FBTR. This presentation described the completion of 29 irradiation campaigns and three and a half decades of successful operation since its inception. and the various safety up gradations, operating experience with sodium pumps, experience with failed fuel localization, life assessment & extension studies of FBTR, replacement of liquid metal seals in block pile, replacement of Steam Generator modules, one with tube leak and other with shell side sodium leak, modification done in the reactor protection circuit to avoid SCRAM during lowering of control rods (LOR).

G. Muralitharan (India) presented an overview of more than 36 years of fuel handling experience of the Fast Breeder Test Reactor (FBTR) in Kalpakkam. This presentation described the experience gained during operation of Fuel Handling system, details various incidents, their causes, corrective action taken, and improvements made in FH system to improve the availability of the system. So far, 99 Fuel Handling Campaigns have been carried out for altering the core configuration as and when required.

K.G. Legkikh (Russian Federation) presented in detail the experience of operational chemical cleanings (washings) (OCC) of the BN-600 steam generator evaporator modules for the entire period of operation from 1981 to the present. This presentation described the stages of OCC, depending on the composition of accumulated deposits during the inter-washing period, the change in the total amount of washed-out deposits, the OCC effect on the corrosion rate of 10Kh2M steel of evaporator tubes, and the stages of the OCC process.

P. Visweswaran (India) presented the plugging process of the degraded SG tube of PFBR using explosive welding technique by remotely operated device. This presentation covered the design, explosive welding experiments conducted on certain number of samples, non-destructive & microstructure examination of the explosive welds and the qualification of the explosive welded plugs.

A. Izhutov (Russian Federation) presented an overview of more than 52 years of operating experience of BOR-60 reactor, facilities available for irradiation testing of a wide range of fuel, absorber and structural materials and the measures taken for improving the safety and life extension activities of the reactor. This presentation covered various types of fuels used, main experiments conducted, irradiation tests conducted in recent years and the prospects of in the BOR-60 reactor.

V. Smykov (Russian Federation) presented the development & application of the following innovative safe technologies for the processing of radioactive waste from spent alkaline liquid metal coolants (Na, NaK, NaKHg) at the research reactor BR-10.

- Solid-phase oxidation (SPO) with the slag from copper smelting production sites for drained alkali metals.
- Gas-phase neutralization (GPN) of non-drainable alkali metals,
- Liquid metal chromatography (LMC) for separation and neutralization of mercury from alkali metals.

This presentation highlighted the safety of technologies by the virtual absence of hydrogen release while processing the alkali metal RW.

TABLE 33. PRESENTATIONS FROM SESSION 8.1 – SFR COMMISSIONING, OPERATION, AND DECOMMISSIONING

Chairs: M. Thangamani and Masanobu Arai

ID	Presenter	Designating Member State/Organization	Title of the Paper
500	J. Winston	India	Reactor Core Viewing System for the pre-commissioning stage inspection of reactor core components of Prototype Fast Breeder Reactor
513	A. Kumar	India	Experience in Preheating of PFBR Reactor Assembly
514	N. Sahu	India	Commissioning and Operating Experience for Secondary Sodium Systems and is Auxiliaries of PFBR
494	J. Jose	India	Advanced in-situ Calibration and Probe Release Mechanism for PFBR SG Inspection System (PSGIS)
141	D. Villani	France	Treatment of sodium of Superphenix Fast Breeder Reactor
217	M. Thangamani	India	Operating Experience of FBTR
469	G. Muralitharan	India	Fuel handling Experience of FBTR
25	K.G. Legkikh	Russian Federation	Experience of operational chemical cleaning of BN-600 steam generator evaporators from corrosion product deposits
492	P. Visweswaran	India	Design, Experimental trials and Qualification of explosive welding technique for plugging of degraded PFBR Steam Generator tubes
131	A. Izhutov	Russian Federation	BOR-60 reactor operating experience, work on improving safety and extending lifetime
30	V. Smykov	Russian Federation	Problems of decommissioning fast reactors and ways of their solution on the basis of the BR-10 research reactor

4.8.2. Track 8: Poster Session

Seven works were presented in the Poster session in Track 8 dedicated to SFR commissioning, operation, and decommissioning. This session involved very unique topics such as description of ultrasonoscopy acoustic imaging system aimed at enhancing safety of the sodium cooled BN-type reactors power unit during its operation; corrosion hydrogen mass transfer in fast reactor steam generators; analyses using a CFD based computational tool to systematically investigate the possibility for development of circumferential temperature gradient in the reactor vessel of Fast Breeder Test Reactor (FBTR); conceptual framework termed CREATE for effective introduction and deployment of Gen IV reactors and the novel electrical, electronics, and instrumentation systems developed and deployed in the fast reactor fuel reprocessing plants as of many of electrical, electronics, and instrumentation systems operate in the harsh radiation and chemical ambience of the hot cells and their operation and maintenance of the systems pose unique challenges.

TABLE 34. POSTER PRESENTATIONS FROM TRACK 8

ID	Presenter	Designating Member State/Organization	Title of the Paper
29	V. Smykov	Russian Federation	Corrosion hydrogen mass transfer in fast reactor steam generators of the sodium-water type
38	X. Hu	China	Design of experimental scheme for activation method of China demonstration fast reactor
196	D. Lesiukov	Russian Federation	Ultrasonoscopy system “VIZUS” for sodium-cooled BN-type reactors
375	K. Chaturvedi	India	Numerical Investigation of Cellular Convection in the Cover Gas space of Fast Breeder Test Reactor
411	B. Wilcox	Nigeria	New Concepts and Methodologies for the Effective Deployment of Gen IV reactors
498	D. Jagadishan	India	Development of a 15 kg servo manipulator for remote handling applications
499	P. Bhanu	India	Novel Electrical, Electronics and Instrumentation systems for Fast Reactor Fuel Reprocessing Plants

4.9. TRACK 9 – EDUCATION, PROFESSIONAL DEVELOPMENT, AND KNOWLEDGE MANAGEMENT

4.9.1. Session 9. 1 Education, Professional Development, and Knowledge Management

Session 5.1 comprised of five presentations, two from the Russian Federation, one from USA, one from Romania and one from the IAEA. One paper from UK was not presented.

G. Tikhomirov (Russian Federation) presented an overview of the challenges society is facing with respect to the global energy demand. He then presented an overview of Gen IV nuclear energy systems which will be deployable no later than 2030 and offer significant advances in sustainability, safety, reliability and economics. The Proryv project offers a new technological platform for nuclear energy based on principles of natural resources management and eco-friendliness. A closed nuclear fuel cycle with new fast reactors will allow eco-friendly energy production and wastes utilization due to nuclear fuel cycle closure with burning long-lived radioactive fission products in the reactor, purification of radioactive wastes, storage, and geological disposal without disturbance of natural radiation balance. He then presented the MePhI Master graduate programme in ‘’Nuclear power technologies of new generation’’ which aims to train and prepare graduates with expertise and knowledge from neutron physics to closed nuclear fuel cycle (CNFC) economics, so that they have an understanding of all the processes that take place in the CNFC and can contribute to the development of GenIV advanced nuclear energy systems. Practical training will take place at JSC "Siberian Chemical Combine" RIAR while theoretical classes will be held at STI NRNU MEPhI whose advantages are the-proximity of the main sites of the fast neutron reactor plant "BREST-OD-300" under construction. This will allow to combine theoretical training of graduates with practical training directly at the production site.

P. Paviet (USA) presented the series of webinars which have been organized since September 2016 by the GIF Education and Training Working Group. They are offered once a month, through an online platform and shared also on YouTube. Channel. In total 64 webinars have been organized in total since its inception. Some data were presented on the number of viewings, geographical distribution as well as organizations distribution**.** The Pitch your GEN IV competition was organized for the first time in 2021. It was a very successful event with 51 abstracts received and there are plans to continue these events to involve and stimulate more the young generation in advanced nuclear energy systems.

A. Nakhabov (Russian Federation) presented the context in which the Obnisk centre for fast reactors is operating, which includes the Obnisk Institute for Nuclear Power Engineering established by MEPhI in 1954 and the Institute of Physics and Power Engineering ‘Leipunsky’. The training which is provided by the institutes was presented. The main training programmes for engineers is focused on the design and operation of NPPs, the design of nuclear reactors, electronics and automation are focused on training specialists for all three main types of power reactors in the Russian Federation – VVER (pressurized water reactors), RBMK (light water graphite moderated reactors) and BN (fast sodium cooled reactors). Starting from the third year they specialize in Fast Reactor technologies. Finally, the students have the opportunity to train on a NPP simulator which can be configured for all three types of reactors (VVER-1000, RBMK-1000 and BN-600) as well as to do some practical training. A Master Programme on Physic and Technology of Fast neutron reactors was established in 2011 for training specialists in fast reactor operation and design, for NPP and research facilities as well. They wish to expand these education and training on FR to attract more foreign students and bring these technologies to the international market.

M. Apostol (Romania) presented first an overview of the ALFRED infrastructure which consists of its R&D activities, the LFR experimental facilities, the Hub and the ALFRED Demonstrator. The investigation on human resources needs for ALFRED implementation was performed in the framework of PRO ALFRED project (September 2019 – November 2020, coordinated by RATEN ICN and funded by Romanian Ministry of Research). The needs (in terms of jobs number) were identified for all the different components of the ALFRED infrastructure and for each identified job, the specializations and the minimal competences were established. Around 600 jobs were identified as needed for the operation of ALFRED demonstrator and its support infrastructure, the hub, and for the R&D related activities. The staff hired through competition will definitely follow a training course that will include at least the following elements: LFR technology, lead and cover gas chemistry, LFR thermal hydraulics, ALFRED infrastructure equipment knowledge, work procedures. The presentation presented also an overview of the educational programmes and Universities, in particular University Politehnica of Bucharest (UPB) and University of Pitesti (UPIT), which will provide the required education and training opportunities. An analysis of the gaps between the actual educational offer and the minimal competences required by the Alfred infrastructure. To cover the gap between the existing competences and specializations provided by UPB and UPIT, the introduction in the current curriculum of new information or disciplines related to LFR technology, the revision of the current disciplines and / or the introduction of new disciplines in the current curriculum, training based on the academic education received in universities and focused on LFR technology, developing of a master programme dedicated to LFR technology as well as the correlation of the research activities to be performed in the six experimental facilities (ATHENA, ChemLab, HELENA2, ELF, HandsOn, Meltin'Pot) with the doctoral programmes**.**

J. Mahanes (IAEA) presented the main activities of the IAEA on Fast Reactor Technologies, including Coordinated Research Projects (CRP), Benchmarks, Working Groups, Technical Meetings, Training courses and workshops. In particular a number of CRPs were presented in more detail together with their past or upcoming publications (TECDOC). The IAEA is currently developing a sodium cooled fast reactor educational simulator with the help of an external vendor. It is a pool type design with 600 MW(e) capacity. The simulator will have a well-developed graphical user interface to make it easier for teaching and learning purposes. The emphasis is to explain the characteristic features of a typical sodium cooled fast reactor and to improve the understanding of physics and technological aspects of these type of reactors. The simulator is intended only for education and training purposes and cannot be used for safety analysis or detailed research and development. The simulator is under development and should be ready by 2022. Other activities of relevance to the community planned for 2022 are a training course with Basic Principles SFR Simulator as well as a Regional WS on Advances in the Modelling and Simulation of Thermal Hydraulics in Liquid Metal Cooled Fast Reactors, India.

TABLE 35. PRESENTATIONS FROM SESSION 9.1 – EDUCATION, PROFESSIONAL DEVELOPMENT, AND KNOWLEDGE MANAGEMENT

Chair: Antonella di Trapani			
ID	Presenter	Designating Member State/Organization	Title of the Paper
167	G. Tikhomirov	Russian Federation	Training of new generation specialists in the field of fast neutron reactors and nuclear fuel cycle closure
437	P. Paviet	USA	GEN IV International Forum webinars initiative
201	A. Nakhabov	Russian Federation	Preserving and transferring knowledge in the field of fast reactor technologies. Experience of the Obninsk Institute of Nuclear Power Engineering MEPhI
424	M. Apostol	Romania	Investigation on Human Resources Needs and Competences Building for ALFRED Implementation in Romania
540	J. Mahanes	IAEA	Overview of IAEA Fast Reactor Related Technology Development Activities

4.9.2. Track 9: Poster Session

No Posters were presented.

5. SUMMARY OF PANEL SESSIONS

5.1. PANEL I – INNOVATIVE FAST REACTORS: DESIGNS, APPLICATIONS, AND FUEL CYCLES

This panel focused on the role of fast reactors and fast SMRs in decarbonization of energy production. Five experts from different Member States presented and discussed the latest innovation in the respective countries and organizations, presented national and private fast reactor programmes and reactor designs, and discussed with the audience start-ups vs. governments supported projects and analytical and simulation capabilities on design organization and national regulators. This panel was moderated by Mr. V. Kriventsev, IAEA.

P. Gauthé, Project manager for a R&D project dedicated to new innovative reactor sketches of SFR and MSR types at Commissariat à l'Energie Atomique (CEA), France
J. Latkowski, Senior Vice President of innovation at TerraPower, USA
S. Raghupathy, Head of Reactor Design & Technology Group and Electronics & Instrumentation Group as Group Director in Indira Gandhi Centre for Atomic Research, Kalpakkam, India
N. Mosunova, Head of department, Nuclear Safety Institute of the Russian Academy of Sciences (IBRAE RAN), Russian Federation
X. Huo, Senior Research Engineer, Department of Reactor Engineering, China Institute of Atomic Energy, (CIAE)

P. Gauthé presented the major questions for innovators in nuclear domain: Besides large PWR, what kind of reactors are necessary for the deep decarbonatization (Net Zero 2050) and energy sustainability? To answer that, there are two main drivers for innovation: One is deep decarbonization with new uses o nuclear energy such as heart, H_2, desalination, small grids. High pace on announcements on advanced reactors, the issues which are underestimated are fuel supply, licensing, and waste. The second key driver for innovation is sustainability. For efficient use of resources and waste minimizations, the FR are strategic and necessary for public acceptance. Sustainability should not focus only on nuclear fuel but also other raw materials such as Beryllium, Hafnium, Lithium and Helium which are becoming more and more expensive. At CEA, the focus of innovative projects is on Small Modular Reactors for short and middle term markets, and advanced modular reactors such as fast reactors for sustainable energy and new uses of nuclear heat and waste management focused on molten salt reactor and transmutation.

J. Latkowski introduced two TerraPowers reactor projects: the molten chloride fast reactor developed under two DOE programmes (Advanced Reactor Concepts and Advanced Reactor Demonstration Programme) and NATRIUM sodium fast reactor with segregation of nuclear and non-nuclear side projected to drive down the capital cost and also coupling with molten salt thermal storage giving the plant ability to integrate with renewables and load follow.

S. Raghupathy presented the overview of the (Fast Breeder Test Reactor) FBTR-II., loop type sodium cooled reactor design. Innovativeness of this reactor, in the Indian context, is due to the metal fuel with high breeding ratio. In India, the reactors need to be deployed on large-scale to meet the energy security. The high breeding ratio aspect is needed to meet the net zero requirements and effectively substitute the fossil fuel powered plants in India. The aim of the FBTR-II reactor is to gain experience running the reactor with metal fuel and address the safety aspects of metal fuel. The FBTR-II will meet all R&D requirements and provide required continuity for the programme with the major aim to close the fuel cycle.

N. Mosunova briefed on the innovative fast reactor designed in Russian Federation. Target requirements for commercial reactors with lead and sodium coolants were mentioned, such as removal of restriction on fuel resources due to efficient use of uranium raw materials during multiple recycling of nuclear fuel in the closed NFC, economic competitiveness with other energy sources, closing the NFC, elimination n of severe accidents with radiation consequences requiring evacuation and technological enhancement of the nuclear nonproliferation regime.

Two innovative reactors currently developed in Russian Federation are BN-1200 with sodium coolant and BR-1200 with lead coolant. Regarding the lead coolant, design concept of BREST-OD-300, the demonstration reactor was presented. It involves features such as mixed nitride fuel, integral layout, no shutoff valve in the primary circuit and passive emergency cooling system. The BREST-OD-300 construction started in 2021. All elements of closed nuclear fuel cycle should be demonstrated at the pilot demonstration energy complex (PDEC). The construction and commissioning of the fuel refabrication module should be finalized in 2023 while the reprocessing module will be finalized by 2029. At the end, the evolution of the operating prototypes BN-600 and BN-800 reactors was also summarized highlighting the advantages of the nitride fuels and stating that the sodium cooled technology is ready for commercialization.

X. Huo introduced the prospects of next generation fast reactor of China and presented the timeline of fast reactor development in China. The demonstration reactor CFR600 is under construction and is preparing for commissioning in 2023. He highlighted the main considerations for next generation FR as the following three features: key technologies to realize sustainability such as advanced closed FC, related pyro processing for full actinide recycle; key technologies to realize the economy such increased reactor power, high burnup and large steam generators and key technologies to realize safety such as passive residual heat removal and elimination of off-site emergency. Another option to sodium-water power conversion is the supercritical carbon dioxide which is under research. In China, the commercial fast reactors are expected to be deployed in 2030s.

The discussion touched on many topics and aspects, such as safety analysis and the role of designers vs. regulators. Participants mentioned that regulator should follow the rule of reasonable caution, both should be experienced but personal qualities should be different, the designer creates new facilities while the regulator critically considers other solutions. Other topics included the possibility and challenges of deployment of fast reactor in countries with no prior nuclear programmes, the comparison of nitrides and metallic fuels and role of private company funding in accelerating the reactor developments and deployments. In many countries the concept of combination of private sector money and government funding is common. Role of SMRs in decarbonizations efforts was highlighted by participants as a way to decarbonize with fission energy that cannot be decarbonize by large PWRs. SMRs and micro reactors are key for such grid security, industrial heat, H_2, green ammonia and others. Finally, participants agreed that in order to get public acceptance, there is a need to demonstrate nonelectric application of nuclear reactors. It was also concluded that small innovations of current designs may not get appreciated if they are not finalized for deployment.

5.2. PANEL II – STRENGTHENING FAST NEUTRON SYSTEMS' COMMUNITY: EMPOWERING THE NEXT GENERATION'S PROFESSIONALS, TOWARDS GENDER BALANCE, CROSS-CUTTING DISCIPLINES

Fast reactors are a vital part of the history of nuclear power, and their strengths of sustainability, efficiency, and safety have never been more important to the world. Many are attracted to this field due to the incredible energy economy of fast reactors, the possibilities of waste transmutation and management, and the promise of a closed fuel cycle. Many others are attracted to the field of advanced reactors due to the unique challenges in materials or in modelling and simulating liquid metal coolants.

Technology is not only the understanding of the physical systems, but technology can also refer to how one thinks and communicates that understanding. This conference is held not only to discuss the status and advancements in fast reactor and related fuel cycles, but also how to strengthen, attract, and prepare the next generation of fast reactor professionals. It is vitally important to grow the field of both nuclear power and advanced reactors to meet both climate change mitigation goals and energy need projections. This panel focused on three opportunities to grow the field: striving towards gender balance, supporting early career professionals, and attracting experts from other fields.
Five panelists, experts in their field, hosted a discussion between panellists and the audience assessing where the field of advanced reactors stands in regard to each of the three topics, as well as personal stories and strategies for success. The panel was moderated by Mr. J. Mahanes, IAEA.

A. Bachrata, Researcher at CEA; Lead of WiN Global Young Generation Group, CEA, France
J. Ramesh Mote, Reactor Design & Technology Group, IGCAR Kalpakkam, India
A. Di Trapani, Head of the NEST and Global Forum Secretariat and Senior Analyst for Education, Outreach and Knowledge Management OECD Nuclear Energy Agency (NEA)
A. Duncan, Deputy Assistant Secretary for International Nuclear Energy Policy and Cooperation, DOE-NE, USA
E. Hamase, Reactor Core and Plant System Evaluation Group, Fast Reactor Cycle System Research and Development Center JAEA, Japan

The panel identified opportunities for attracting women and young engineers to nuclear field and advanced reactors areas. It strongly illustrated the importance of women role models, building of long-term mentor/mentee relationships and enabling flexibility. Panellists also shared innovative ideas for widening the advanced reactors field with digital technologies, machine learning, use of Artificial Intelligence tools, policy advocacy and other means for attracting diverse skills. The importance of increasing the number of women in STEM fields, aligning organization's expectations and personal goals, encouraging curiosity in junior team members and willingness to grow beyond competitiveness was highlighted. Finally, spreading the attraction of working in the nuclear field to women students at the early stage was mentioned by the panellists as one of the key tools to attract more talents to nuclear and advanced reactors field.

6. SUMMARY OF YOUNG GENERATION EVENT

6.1. YOUNG GENERATION ESSAY CONTEST

The United Nations 2030 Agenda for Sustainable Development and the Sustainable Development Goals (SDGs), adopted by world leaders in 2015, aim at transforming the world by eradicating poverty and moving toward sustainable development. Achieving these goals will require reliable and sustainable sources of energy. In the framework of the IAEA International Conference on Fast Reactors and Related Fuel Cycles, the Organizing Committee invited early career professionals to make proposal on how to address the following United Nations Sustainable Development Goals (SDGs), while discussing the necessary innovative technology and methods.

FIG. 1 United Nations Sustainable Development Goals (SDGs)

In the challenge, participants were asked to submit a short essay (1500 words maximum) on emerging ideas and innovations related to fast reactors nuclear technology and related fuel cycles. The paper should:

- Identify a novel or emerging technology related to fast reactors or an idea for future work
- Identify where it may be deployed or where it may have the most impact
- Identify challenges to adopting the technology, and opportunities for future research

The objective was to provide young professionals (below 35 years old) with a space to acknowledge and discuss new strategies, technologies, tools, and ideas.

6.2. YOUNG INNOVATOR WINNERS' PLENARY PRESENTATIONS

6.2.1. Small Modular Fast Reactors for the ASEAN Region: Implementation of the TRISO Fuel Particle Concept as a Regional Variant of the Fast Reactor - Tan Zhe Chuan

Nuclear power has always been contemplated by many ASEAN (Association of Southeast Asian Nations) member countries, but never fully implemented. Despite being a viable source of clean energy, South East Asian counties face multiple challenges in introducing the technology into the region; whilst some are technical in nature, others are political, making the implementation of a nuclear industry a prickly, multifaceted issue to tackle.

Technical challenges plaguing the region can be summed up into two main categories: 1) Lack of Technical Expertise and 2) Disaster Prone Region. Most South East Asian countries collaborate with other existing nuclear nations such as China or Russia (Bunthoeun, 2020; Shin, 2016; Archana Rani, 2019) to tap on the technical expertise of the partner nations, given a considerable lack of local expertise familiar with nuclear engineering and nuclear power plant implementation, design, operation, and crisis management. The build-up of local human resources in the South East Asian countries can take years to cultivate.

In addition, many South East Asian countries are located in disaster prone regions and earthquake fault lines. For example, Philippines and Indonesia lie on the Pacific Ring of Fire, where 90% of the world's earthquakes are prone. After the 2011 Fukushima Nuclear Incident, where a nuclear power plant in Fukushima, Japan, experienced a Level 7 accident after being struck by the Tōhoku earthquake and tsunami, disaster related concerns regarding nuclear energy in South East Asia are abound and pervasive.

This also ties in with the geopolitical views on nuclear energy- a large majority of citizens in South East Asia view nuclear power unfavourably, especially since the 2011 Fukushima Incident. These views are common across the region, such as in Singapore (Ho, 2018) and Indonesia (Wisnubroto, 2019). The most common reasons for their unfavourable views include their lack of understanding of nuclear power plant systems, their worries regarding nuclear power plant-related disasters, as well as fears of mismanagement and corruption during nuclear power plant crises.

Despite the challenges involved in implementing nuclear power in the region, the attractiveness of nuclear cannot be underestimated in the region. With a population of more than 680 million people, South East Asia has nearly the same population size as Europe. In many member states, the unique topography makes the implementation of traditional Light/Heavy Water Reactors unfeasible, such as with maritime South East Asia, where the population is spread across 25,000 islands. Small Modular Fast Reactors (SMFRs) provide a possible avenue for nuclear power implementation in the region, with high burnup and sustainability, inherent safety and greatly reduced waste production. The demand for nuclear power in ASEAN cannot be underestimated nor ignored.

Current research on small modular fast reactors has been greatly focused on the feasibility from the core design perspective. Such work revolves mainly around the optimization of the core design, and couple 3D computations (Peakman et al, 2018). Work done includes areas such as thermo-hydraulics (Zhao et al, 2020; Wei et al, 2019) and neutronics (Guo et al, 2022; Wei et al, 2019).

However, despite the large amount of research being poured into SMFRs, a gap exists in terms of analysing SMFRs from the perspective of Beyond Design-Basis Accidents (BDBAs). Current literature on safety analysis includes research on natural circulation, safety features, and passive cooling (Guo et al, 2021; Zhao et al, 2016), but explicit research to determine the effectiveness of such safety systems under BDBA scenarios is lacking compared to other Gen IV reactor designs (IAEA, 2021). For countries in or neighbouring disaster prone regions, especially in South East Asia, the Small Modular Fast Reactor does not hold up as strongly compared to its counterparts under a clinical analysis of the costs and benefits, yet the sustainability behind its technology is attractive to many. Research in the areas of BDBA scenarios and corresponding SMFR safety will improve the attractiveness of SMFRs to this region.

In the process of analysing the SMFR from a safety perspective, a potential solution may come in the form of adopting the fuel type of the High Temperature Gas cooled Reactor (HTGR): Tri-structural Isotropic (TRISO) fuel particles. This paper proposes exploring the potential combination of TRISO particles, as well as potentially utilizing the concept of a pebble-bed reactor in the fast reactor. While the concept of 3 | P a g e the pebble bed fast reactor has been previously explored (Kouichi & Hiroshi, 2001), the possibility of implementing TRISO fuel in SMFRs has not been fully explored. The traditional TRISO particle used in HTGRs is not suitable for fast reactors due to low heavy metal densities (Meyer et al, 2007), but the concept behind the TRISO fuel particle can be adopted and refined to meet the heavy metal density requirements whilst exhibiting safety characteristics such as withstanding extreme temperatures and fission product retention. Furthermore, research on the usage of the TRISO design with reactor-grade plutonium in the CANDU reactor (Şahin, 2010) has shown that the TRISO particle concept and design has immense potential in terms of meeting the constraints posed by the regional demands in South East Asia at the cost of certain reactor efficiencies. This modification of SMFR can allow for feasible regional variants for disaster prone and/or island nations.

There are major challenges involved with this potential solution; namely, by virtue of revamping the fuel design, data related to thermal hydraulics, neutronics, safety analysis, etc. needs to be re-analysed and researched on. The redevelopment of the fuel type may also invalidate current proof-of-concept experiments, with new test reactors required to determine the feasibility of the concept.

The demand for nuclear energy in ASEAN is rising, and many countries in the area has begun to take steps to develop nuclear expertise in the region. As such, preferences for nuclear reactor design in the region are mild and almost non-existent. This presents an opportunity to introduce the fast reactor as a sustainable, effective reactor; however, the fast reactor concept has its flaws as well. This paper proposes exploring the modification of the fuel type in the reactor to meet the unique demands and requirements of the South East Asian region. By adopting TRISO fuel particle design concepts, an improved fast reactor variant may be developed for the ASEAN region.

6.2.2. Production of Mo-99 isotope in the BN reactor by beryllium blocks - Kucheryavykh Oxan

Some of the radionuclides that are produced in the reactor, like ^{235}U fission fragments, are quite important. For example, the isotope ^{99}Mo is widely used in medicine to make radiopharmaceuticals used in diagnostic procedures and treatment of cancer. But it is not the molybdenum-99 isotope itself that is used in medicine, but the short-lived technetium isomer - technetium-99m (^{99m}Tc). It is a by-product of β-decay.

For example, the total number of diagnostic procedures worldwide based on the most widely used radionuclide ^{99}Mo is currently estimated at 25-30 million per year with a growth rate of 1.5%-2.5% over the next ten years. The supply of ^{99}Mo will remain critical over the next decade, as techniques and equipment using ^{99m}Tc are now used in the great majority of nuclear medicine procedures.

Molybdenum-99 is produced at nuclear reactors in South Africa, Belgium, Holland, Australia, Russia and other countries. However, demand for molybdenum-99 in the world is growing every year.

Global demand for Tc-99m is expected to grow by an average of 3-10% per year as new markets develop nuclear medicine and existing markets continue to use it. The problems of the molybdenum crisis are also related to the shutdown of ^{99}Mo production at the world's leading 99Mo rectors. Therefore, there is a need to produce ^{99}Mo.

This essay proposes the idea of producing medical isotope 99Mo using the BN-600 fast-neutron reactor.

In order to keep large fluxes and soften the spectrum, we use beryllium blocks. Since beryllium is a good neutron moderator, replacing some fuel assemblies with beryllium blocks will soften the spectrum in the target regions.

Fast reactors are known to have a fast neutron spectrum, and the fluxes in them are much larger than in thermal reactors. The question arises why we should consider softening the neutron spectrum in a fast reactor. Since the 99Mo production is determined by the flux value and the ^{235}U fission cross section (these values should both be large enough), there is a need to make the thermal spectrum on the one hand, and to keep the neutron fluxes inherent in fast reactors large enough on the other.

Nuclear reactors mainly use the uranium-235 – $^{235}U(n,f)$ ^{99}Mo fission reaction to produce molybdenum-99, which has a cross section of 582.6 barns.

The beryllium block will be a structure that has the same size as the fuel assembly. The block will contain an isotope production target, a target cladding, and beryllium surrounding the target. Enriched uranium dioxide can be used as targets. Aluminium alloy is used as the cladding for such a target in standard VVRc, but in the BN-600 reactor, the aluminium alloy may melt under the influence of high temperatures. This problem could be solved by using a titanium alloy.

6.2.3. Advanced Functional Materials for Next-Generation Fuel Reprocessing - Kuntal Kumar Pal

Partitioning and transmutation (P&T) strategy is inevitable for the safe management of high-level liquid waste (HLLW) [1], public acceptance of nuclear power and sustainable growth of nations. A good waste management practice dictates the policy makers and researchers to develop technologies to minimize or eliminate the presence long-term radioactivity in the environment and avert the hazards associated with radiotoxic elements to the future generation. This can be achieved only by partitioning radiotoxic elements present in nuclear wastes and transmute them into stable or short-lived products in fast reactors or accelerated driven systems. In addition, the nuclear wastes also contain certain radioactive metals that are of societal importance. Separation of these metals could offer the advantage of using them for societal applications, especially in the health sector, such that the importance of nuclear energy is realized by the public.

HLLW arising from fast reactor fuel reprocessing is composed of long-lived radiotoxic actinides (^{241}Am, ^{243}Cm) associated with chemically similar lanthanides, long-lived radiotoxic fission products (^{137}Cs, ^{90}Sr), other fission products and corrosion products in 3-4 M nitric acid medium. The separation of the trivalent actinides and long-lived fission products from HLLW is a challenging job owing to the presence of chemically similar lanthanides and higher concentration level of nitric acid (3-4 M). In addition, these actinides and fission products are present in very small concentrations making the separation even more difficult. All these factors call for the development of advanced methods and materials for separations. Currently, the methods available for the separation of trivalent actinides and fission products are based on liquid-liquid extraction [1-3]. These methods suffer from the primary drawback of third phase formation, use of unconventional diluents, presence of phase modifiers in significant concentration, poor selectivity, etc. In addition to this, methods based on solvent extraction generally results in the generation of a large volume of radioactive organic waste. The treatment and disposal of such organic waste incur significant expenditure and require environment benign technology.

Graphene and graphene oxides are one of the unexplored materials for radioactive waste management. These materials have excellent physical and chemical properties amenable for surface modification. The striking features of graphene that make them suitable for radioactive waste management are (a) large surface area, (b) excellent radiation stability due to the presence of extended conjugation, (c) complete incinerable nature, etc. However, graphene as such is not suitable for the separation of a metal from an aqueous solution due to the absence of functional groups that can coordinate with metals. Nevertheless, these graphene and graphene oxides are amenable tuning and organo-modification over their surface, which can be exploited for advanced task specific separation.

Towards the development of advanced functional materials, the primary goal of the research proposal is to link the chemical functionalities, responsible for task specific metal capture, covalently to the basal plane of graphene sheet. Such ligand functionalized graphene is then suspended directly in aqueous waste solution for capturing the targeted metal and separate the suspended graphene from aqueous solution by mechanical operations, such as filtration, centrifugation, etc. For example, the diglycolamide functionalized graphene material can be employed for capturing lanthanides and actinides selectively from the rest of the fission products present in the PUREX raffinate (Fig. 2). Similarly, the other ligands such as triazine derivatives, crown ethers, diethylenetriamine pentaacetic acid etc., can be attached to graphene sheet for targeted separation of trivalent actinides and fission products. In the next step, the captured metal can be recovered from the solid phase by a suitable aqueous medium. The aqueous product obtained can be processed further to obtain them in required form and the solid/dispersed material can be recycled for the next batch. Moreover, the functionalized graphene material can be drawn as a sheet form for separations based on membrane technology.

Like any other material employed for nuclear applications, the functionalized graphene also undergoes chemical and radiolytic degradation during processing of nuclear waste solution. The computational calculations can be explored to elucidate the optimum chain length between the basal plane of graphene and the ligand, and to deduce the structure of the metal complex to bring out the insights of structural properties. Such studies serve as inputs for minimizing the radiolytic degradation and for the improvement of selectivity and separation efficiency. Homogeneous dispersion of these functional graphene materials in emulsions and in certain organics could help in developing advanced liquids for cleaning of radioactivity contaminated surfaces. The dispersion of graphene in organics can be facilitated by suitably adjusting the carbon to oxygen ratio of the graphene sheet. The functionalized and exfoliated graphene sheet emulsions can be painted on the contaminated surface and peeled from the surface for decontamination.

In view of the above, it is obvious that graphene based functional materials can play a vital role in mitigating many of the challenges posed during fuel reprocessing and waste management. Transmutation of actinides obtained after partitioning eliminates the hazards associated long-lived actinides making the nuclear power acceptable to the public and the environment benign. In addition, the separation methods based on graphene could minimize the generation of radioactive waste. Graphene based functional materials can also be employed for the separation of the fission products for societal applications in a similar fashion. Since graphene are being made up of carbon, oxygen and/or nitrogen atoms, they are completely incinerable which makes them the attractive option for the treatment of nuclear wastes in the near future.

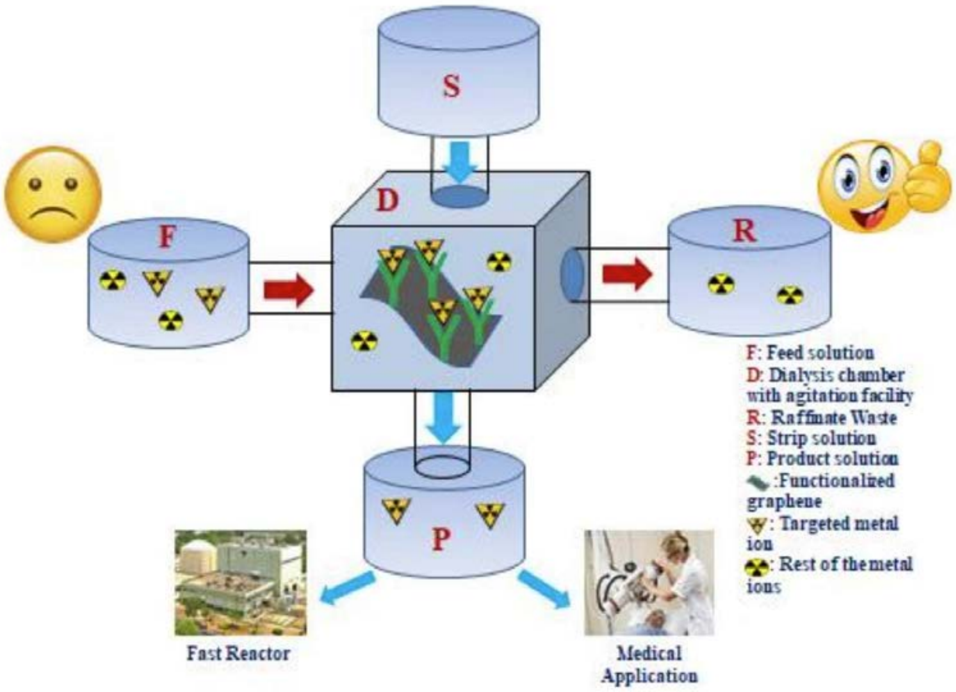

FIG. 2 Metal ion separation with functionalized graphene. F: Initial Feed solution; D: Dialysis chamber with agitation facility; R: Raffinate; S: Strip solution; P: Product solution.

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7. CLOSING SESSION

7.1. CHAIR OF THE INTERNATIONAL ADVISORY COMMITTEE

Arun Kumar Bhaduri

Chair, International Advisory Committee, Closing Summary

A very good evening, on behalf of all of you here, online and in the audience, I would first like to compliment the Agency for a wonderfully done conference. Coming to the general report on technical and plenary sessions, I would like to by name thank the DDG Mikhail Chudakov, Amparo, Vladimir, Joseph, Nikoleta, Alina and Krzysztof for their untiring efforts during these four days and of course the Director General for inaugurating this important conference and we believe that this will be a trend setter for future conferences.

With around 300 technical contributions and 500 participants, the international conference on fast reactors and related fuel cycles (FR22) attracted participation from all over the world. In average, about 50% of the accepted abstracts contributors have submitted full papers. Overall, a total 25 oral technical sessions with 198 presentations, three plenary sessions with eleven keynote presentations a poster session with 65 presentations were organized. 42 participants submitted contributions to the Young Generation Event Essay contest with 3 winners selected by the ISCP committee, who then had the opportunity to present their winning ideas during one of the plenary sessions. The two thematic panels: Innovative Fast reactors: Designs and Applications and Strengthening Fast Neutrons Systems' Community were well organized and very thought-provoking. From 542 submitted abstracts, 288 papers were submitted with majority of papers submitted from Russian Federation, United states of America, India, China and France. Table 35 below provides the titles of each technical track.

TABLE 36. TECHNICAL SESSIONS OVERVIEW

Plenary Session	National and international fast reactor programs
Track 1.	Innovative Fast Reactor Designs
Track 2.	Fast Reactor Safety
Track 3.	Fuels, Fuel Cycles and Waste Management
Track 4.	Fast Reactor Materials (Coolants, Structures) and Components
Track 5.	Test Facilities and Experiments
Track 6.	Modelling, Simulations and Digitalization
Track 7.	Sustainability: Economics, Environment and Proliferation
Track 8.	Commissioning, Operation and Decommissioning
Track 9.	Education, 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 several international organizations: China, France, India, Japan, Republic of Korea, Russian Federation, United States of America, European Commission, 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, especially since FR17. Considering the thematic tracks, the one area we are community need to concentrate more is the newly added track 9 on Education, Professional Development and Knowledge management to increase the submissions in this topic.

The highlights during the plenary discussion on national and international programmes include development of new national infrastructures to support FR and FC demonstration, need for harmonization and standardization through ASME, importance of support and fostering the private-public partnership and importance of international collaboration through international organizations and their coordination. The panel participants and keynote speakers have advised the IAEA to establish the Fast Reactor and Fuel Cycle platform, similar to SMR platform. This has been supported by FR22 delegates.

In Track 1 on innovative fast reactor designs, out of 31 accepted papers, 23 were selected for oral presentation and 8 for poster presentation. The track was divided into three technical sessions.

TRACK 1. INNOVATIVE FAST REACTOR DESIGNS

Session	Title
1.1	Overviews and Fundamentals of Fast Reactors
1.2	Innovative Design Advances
1.3	System Innovations

The presentations provided discussion and information exchange related to experience and techniques from reactors currently in operation or design of future reactors in various Member States including diverse reactor types. Great variety in size, mission and maturity of the fast reactors presented ranging from conceptual designs to a plant under construction, and from reactors envisioned for training/education purposes to research reactors and power reactors. Innovation in design solutions and materials is pursed for continuous improvement and step-changes in performance and reactor mission. Global overview of different fast reactor designs has been given covering large part of the world geographically and covering different types of coolant types. Experimental activities, both already performed and planned, continue to play a key role in substantiating technical performance and safety case of fast reactor technologies.

The track provided great overview of the developments of the BN Series of Reactors in Russian Federation, the progress of future PBRs in India, the US Fast Reactor Technology R&D programme, the progress in the Japanese Sodium Cooled Designs, the lead fast reactor activities in GIF, the progress of the ALFRED Lead cooled reactor to be built in Romania, the recent advances in organization and in design of the MYRRHA ADS programme in Belgium and The Gas cooled Fast Reactor R&D activities in |Central and Eastern Europe.

In Track 2 on fast reactor safety, out of 39 accepted papers, 32 were selected for oral presentation and 7 for poster presentation. The track was divided into four technical sessions.

TRACK 2. FAST REACTOR SAFETY

Session	Title
2.1	General Safety Approach
2.2	Safety Design and Analysis
2.3	Accident Analysis
2.4	Severe Accidents

Various presentations in this track encouraged useful discourse on the safety information and knowledge of fast reactor. Member States have had time to share various information on areas such as safety approach design, analysis and assessment of Generation IV fast reactors: development and numerical evaluation of safety systems such as passive decay heat removal systems, core catchers, fire hazards protection system. Accident analysis covered a range of not only modelling but also neutronics and thermal hydraulics calculations and source term simulation, which provided an opportunity to identify the development stages and status of experimental programmes of SFR accidents and reorganize future research directions.

In Track 3 on fuels, fuel cycles and waste management, out of 42 accepted papers, 29 were selected for oral presentation and 13 for poster presentation. The track was divided into eight technical sessions.

TRACK 3. FUELS, FUEL CYCLES AND WASTE MANAGEMENT

Session	Title
3.1	Fuel Cycle Scenarios
3.2	Development of Innovative Fuels: Design and Properties Irradiation
3.3	Reprocessing, Partitioning, and Transmutation
3.4	Advanced Fuel Development

This track was an opportunity to share technical information of Member States and approaches to closing the nuclear fuel cycle using fast reactors. In particular, speakers discussed various aspects of fuel cycles and waste management from insights on the project i.e., EU PUMMA on Pu management, Advanced Reactor Experiments for Sodium Fast Reactor Fuels (ARES) Project, fuel options for Generation IV reactors, characterization and fabrication of fast reactor fuels, reprocessing of spent nuclear fuel for recycling and transmutation, partitioning and actinide burning using fast reactors. This track also suggested future development direction through providing exchange of useful information.

In Track 4 on fast reactor materials (coolants, structures) and components, out of 21 accepted papers, 14 were selected for oral presentation and 7 for poster presentation. The track was divided into two technical sessions.

TRACK 4. FAST REACTOR MATERIALS (COOLANTS, STRUCTURES) AND COMPONENTS

Session	Title
4.1	Advanced Reactor Cladding and Core Material, Coolants, and Related Chemistry
4.2	Structural, Novel, and Large Component Materials

In this track, topics covered focused on material properties and behaviour, novel methods on manufacturing reactor components such as Additive Forging Technology and review of the processes developed to support the safe and reliable operation of sodium system, including sodium purification and advanced fuel handling system. Presented contributions covered studies carried out China, Russia, Saudi Arabia, USA, India, Japan, France and by EC. The overview of studies dedicated to creep and tensile properties of Indian advanced fast reactor clad tubes for future FBRs and irradiation induced damage in SS 316 (LN) and SS 304 (LN) steels. Studies carried out in China on the fabrication and performance assessment of 14 Cr FeCrAl cladding tube material. Also, studies dedicated to the development of diffusion-type hydrogen meters (DTHMs), used to detect water ingress in the secondary sodium in USA. Studies on creep and creep-fatigue behaviour of the advanced stainless-steel alloy 709 in Saudi Arabia. An overview on the material data acquisition activities carried out by the Japan Atomic Energy Agency (JAEA) to develop the material strength standard for Sodium cooled Fast Reactors (SFRs).

In Track 5 on test facilities and experiments, out of 29 accepted papers, 24 were selected for oral presentation and 5 for poster presentation. The track was divided into three technical sessions.

TRACK 5. TEST FACILITIES AND EXPERIMENTS

Session	Title
5.1	Experimental Reactors and Facilities
5.2	Experimental Program I
5.3	Experimental program II

In the technical sessions pertaining in this topic, papers covered a number of existing and planned experimental facilities. R&D test facilities on sodium fire, plenum thermohydraulic, heat exchanger, and instrumentation, were discussed. Theses phenomena are recognized as important for development and licensing of advanced reactors. Test reactors to support fast reactor development, and experimental facilities to support the development of those reactors and support validation of modelling tools. Presentations were received on important test reactors and facilities supporting them, such as VTR, MBIR, ALFRED, BREST 300 OD, FKBN-2, and other still. Presentations on experiments largely focused on enhancing the safety and flexibility of fast reactor systems. Future experimental programme capabilities, priorities, and needs were discussed by participants in the Q&A forums.

In Track 6 on modelling, simulations and digitalization, out of 57 accepted papers, 43 were selected for oral presentation and 14 for poster presentation. The track was divided into six technical sessions.

TRACK 6. MODELLING, SIMULATIONS AND DIGITALIZATION

Session	Title
6.1	Neutronics
6.2	Thermal Hydraulics
6.3	Multiscale and Multiphysics Calculations
6.4	Simulation Tools for Safety Analysis
6.5	Integrated Analysis and Digitalization
6.6	Fuel Performance and Material Modelling

These presentations highlighted Neutronic analyses of SFR, LFR, MSFR validating new codes and models and calculations of reactivity feedback coefficients, coolant temperature coefficients and void effects to understand the neutronics behaviour of fast reactors. System, sub-channel and CFD calculations as well as model improvements for fast reactors of different designs. Effect of flow induced vibrations as well as flow blockage analysis in a fast reactor fuel assembly. Evaluation of modelling the heat transfer phenomena is a liquid-metal coolant using a light water test facility. Improved understanding of the mechanics of sodium ejection in case of ULOFA or TOPA is of critical importance for an LMFBR safety.

Several Multiphysics codes are being developed specific to fast reactors owing to different physics that needs to be solved simultaneously for better prediction of phenomena. Further discussion focused on ways to develop and get new code certified by national authors. Development of coupling frameworks that can support explicit or implicit coupling of existing codes will enhance Multiphysics simulation capabilities. Verification and validation of these codes is always challenging, and more experimental data is needed. Challenges in development of two-phase flow models for sodium fast reactors were also mentioned. It was proposed to improve simulation results by coupling codes at different scales. Requirements and best practices applicable to nuclear power plant designers for the development and use of analytical codes and methods used in the design and safety analysis of nuclear power plant as well as use of Artificial Neural Network for predicting physical phenomena in Fast Reactors and making digital twins to predict complex phenomena combining multidisciplinary views were discussed.

In Track 7 on sustainability: economics, environment and proliferation, out of 15 accepted papers, 11 were selected for oral presentation and 4 for poster presentation. The track consists of one session.

TRACK 7. SUSTAINABILITY: ECONOMICS, ENVIRONMENT AND PROLIFERATION

Session	Title
7.1	Sustainability: Economics, Environment, and Proliferation

In this track, technical and economic aspects of commercial sodium fast reactors were the main topics of discussion for the speakers. They covered modelling and optimization of economic structures of global deployment of fast neutron systems. Moreover, there was an opportunity to exchange the information of Member State’s technological support for non-proliferation. They also presented specific features of export potential of fast reactor technologies and future plan of research. This area also needs more campaigning for FR25.

In Track 8 on commissioning, operation, and decommissioning, out of 18 accepted papers, 11 were selected for oral presentation and 7 for poster presentation. The track consists of one session.

TRACK 8. COMMISSIONING, OPERATION AND DECOMMISSIONING

Session	Title
8.1	SFR Commissioning, Operation, and Decommissioning

Presentations encouraged discussion on fast reactor commissioning, operation, decommissioning and development of various inspection tools for Steam Generator. Specifically, contents such as development of reactor core viewing for the pre-commissioning stage inspection of reactor core components of PFBR and advanced in-situ calibration and probe release mechanism for PFBR SG inspection system were dealt with as key topics. Authors from India reported on experience in preheating of PFBR reactor assembly, fuel handling and operation experience as well as commissioning and operating experience for secondary sodium systems and auxiliaries of PFBR. Design changes to improve the system performance and safety were discussed. In the presentation of France, there was overviews of treatment of sodium of Superphenix fast breeder reactor in France. Further presentations included overview of more than 52 years of operating experience of BOR-60 reactor, facilities available for irradiation testing of a wide range of fuel, absorber and structural materials and the measures taken for improving the safety and life extension activities of the reactor.

In Track 9 on education, professional development, and knowledge management, out of 5 accepted papers, 5 were selected for oral presentation and 0 for poster presentation. The track consisted of one session.

TRACK 9: EDUCATION, PROFESSIONAL DEVELOPMENT, AND KNOWLEDGE MANAGEMENT

Session	Title
9.1	Education, Professional Development, and Knowledge Management

Presentations covered a large range of topics on strategies for knowledge management and capacity building. Presentations were received on needs assessment at both the project and national levels, showing observers potential opportunities, and skillsets in high demand in the field of advance reactors. Strategies for capacity building though specialized webinars, coordinated research projects, and educational programmes were presented. An extended Q&A illustrated the urgency of the topic. Discussion centered on specific challenges facing knowledge transfer, details of the strategies presented here, and outlooks for the field of nuclear knowledge management.

The IAEA Special Session included seven oral presentations; one IAEA introduction, four on "Neutronic Benchmark of the CEFR start-up tests CRP" covering various work packages within the benchmark and two on "Benchmark of the FFTF Loss of Flow without Scram test CRP" focused on blind phase results. IAEA/CRP on "Neutronics Benchmark of CEFR Start-up Tests" and "FFTF Loss of Flow without Scram test" gave good opportunities for validation of fast reactor tools of member countries. The unique feature of the CEFR benchmark is the allowing for comparison with measured values in clean CEFR core. FFTF blind phase results show relatively good agreement with measured test data and provide confidence that many participants captured the transient progression of the test well.

Fast reactor and related fuel cycle technology remains a proven option as a sustainable source of energy for many generations to come, preserving natural resources and reducing the burden of generated waste to be disposed of. Fast Reactors with closed fuel cycle can compete as renewable source of energy. Therefore, international cooperation on technology of fast reactors and related fuel cycle is vital. SFR is the most mature technology and now the focus is on enhancing safety and improving economic efficiency. Many new promising technologies are under development: heavy-liquid metal cooled reactor (LFR, lead and LBE), Molten Salt cooled Fast Reactor (MSFRs), Gas cooled Fast Reactor (GFR), especially Molten Salt cooled fast reactors are very promising due to the online reprocessing option and its proliferation resistance, most likely with chloride salts. New types of next generation fuels (nitride, metal) will play a crucial role but due to the long experience we have with the MOX fuels, it is likely that it will take a long time to be replaced.

There was general agreement on contribution of organizing the IAEA conference eon Fast Reactors and Related Fuel Cycles every 4 years, next in 2025. Thank you very much.

7.2. CONFERENCE GENERAL CO-CHAIR

Mikhail Chudakov

DDG-NE, Conference General Co-Chair, Closing Remarks

Dear Chairperson, distinguished delegates, ladies and gentlemen,

I would like to thank you for taking part in the International Conference on Fast Reactors and Related Fuel Cycle: Sustainable Clean Energy for the Future, organized by the IAEA in Vienna.

As you know, this Conference takes place every four years, and this year's event follows the previous editions in Kyoto, Paris and Yekaterinburg. Apart 2022 was originally scheduled to take place last year in Beijing but had to be postponed and relocated amid the global pandemic around the world and organized in hybrid format here in Vienna at the agency with in-person and online attendance.

Nonetheless I would like to extend my sincere personnel thanks to China's Atomic Energy Authority for having offered to host the Conference last year. I would like to mention here and to thank China for proposing to host the next edition of this event in 2025.

As I hold the honor and privilege 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 had an outstanding level of participation for a hybrid event, totaling almost 470 participants and 210 observers, drawn from 53 countries and three international organizations. About 100 people took part in person with 45 participating in groups from three remote conference halls, in Nizny Novograd in Russian Federation, and 15 from a remote hall in Beijing, China. 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 in preparation for this event. We were delighted to have received some 290 technical papers from 25 countries, of which 198 were presented orally and 65 as electronic posters. In addition, eleven keynote speeches and five presentations were delivered to the technical panel on innovative fast reactor system, while 42 contributions were made to the Young Generation Innovator Challenge Essay Contest. The three winning young generation event presentations were delivered at the plenary session on Wednesday.

Five women panellists from France, India, Japan, Nuclear Energy Agency (NEA) of the OECD and the United States of America took part in our session on strengthening the fast neutron system community and part in the next generation professionals towards gender balance and cross cutting disciplines. They discussed challenges in attracting young people and women to the field of advanced reactors and provided important insights on mentorship, role models flexibility, the value of curiosity and questioning, and the concept of paying opportunities forward.

In three plenary sessions, we also had the important opportunity to learn more about the latest status of development of fast reactors and related fuel cycle technology in countries with the active fast reactor programmes: China, France, India, Japan, Republic of Korea, Russian Federation and United States of America. During the plenary discussion on the national and international programmes and visions, panel participants and the keynote speakers advised that the IAEA establish a platform on the fast reactors and related fuel cycles, similar to the recently established Agency wide platform on SMRs and their applications. This advice was supported by the FR22 delegates. Together with the IAEA, three other international organizations; the European Commission, the Generation IV International Forum and the Organization for Economic Co-operation and Development presented their members' visions on the topic.

Tuesday:

China: H. Yang, Vice President of the China Institute of Atomic Energy (CIAE), explained that nuclear energy is an important approach for China to carry out the international commitment for reducing carbon emissions.

France: F. Serre, Deputy of the Nuclear Energy Division at the French Alternative Energies and Atomic Energy Commission (CEA), introduced the “Multiannual Energy Plan” that France remains committed to representing the closed fuel cycle policy until 2040.

India: D. Venkatraman, Director of the Indira Gandhi Centre for Atomic Research (IGCAR), summarized the characteristics of Fast Reactors and explained R&D status of the Fast Breeder Test Reactor (FBTR).

Japan: K. Hideki, Japan Atomic Energy Agency (JAEA) introduced “The 6th Strategy Energy Plan of Japan” that was approved by the Cabinet in October 2021: it states utilization possibility of nuclear energy for carbon neutrality.

Republic of Korea: C. Lim, Korea Atomic Energy Research Institute (KAERI), informed that Korea Government has an interest to Sodium cooled Fast Reactor (SFR) coupled with pyro-processing.

Russian Federation: V. Pershukov, Special Representative of ROSATOM for International and Scientific Projects, delivered that New Technology Platform (NTP) of Russia consists of two major elements: Fast Reactor (FR) and Closed Nuclear Fuel Cycle technologies (CNFC).

United States of America: A. Caponiti, Deputy Assistant Secretary of Office of Nuclear Energy in Department of Energy, stressed that advanced reactors are crucial for achieving national and global carbon reduction goals and will be grown by interacting with other energy sources for Net-Zero future.

Wednesday (International Organization)

European Commission: M. Betti, JRC Directorate for Nuclear Safety and Security of the European Commission (EC), explained that JRC tries to assist the advanced reactor research activities. To enhance the design and system integration, many international corporations in Europe are researching new generation nuclear power plants such as Gen IV and the JRC is supporting the growth of these corporations.

Generation IV: R. Hill, Technical Director of Generation IV International Forum (GIF), delivered the GIF Goals and the recent status of recent developments according to the characteristic of each Generation-IV Reactor System.

Organization for Economic Co-operation and Development: T. Ivanova from the OECD Nuclear Energy Agency (NEA) introduced the overview of NEA Fast Reactors activities. NEA has consistently provided a variety of technical reports in order to disseminate scientific knowledge and support development of research.

IAEA: A. Cloizeaux, Director for Division of Nuclear Power of Department of Nuclear Energy of the IAEA, delivered that long-term development of sustainable nuclear power will require fast reactors as they that can effectively utilize natural uranium.

Ladies and Gentlemen, this past week has offered us very exciting opportunities to explore the development of fast reactors and related fuel cycles. We have had the chance to listen to extremely informative presentations and have had the numerous in-depth discussions at the panel and poster sessions.

In particular, the scientific contributions summarized by A. K. Bhaduri in his concluding report on the technical sessions have all exhibited significant technical expertise and demonstrated the vitality and innovation that characterizes this exciting field.

I would like to express my thanks to all of you and especially A. K. Bhaduri for his support as General Chair of the Conference and for his engagement throughout the entire conference.

Of course, I also want to highlight the enormous effort made by the International Scientific Programme Committee, which, under the Chairing of Mr. D. Zhang and Ms. A. des Cloizeaux evaluated over 530 abstracts and then reviewed and finally accepted 288 technical papers.

I also want to thank the chairs of the Panel Session, Ms. Amparo Gonzalez Espartero and Vladimir Kriventsev and the moderator of the Panel on strengthening our fast neutron systems community, Mr. Joe Mahanes.

Please allow me to also thank all Track Leaders of our 9 Technical Tracks and, of course, the Chairpersons of the Technical Sessions.

Finally, I would be remiss not to mention the latest edition of our Young Generation Event. These future experts have once again sent a very strong message that fast reactors and related fuel cycles are a technology for the future. Thanks also to all the 40 plus young professionals who took part in this challenge. And congratulations to the winners from the India, Russian Federation and Singapore.

Both, this youth event and strengthening the community panel underscored not only the importance of young people for the future of innovative nuclear technology but also women, reminding us again why the agency is providing scholarship for female graduate students in the nuclear field through its Marie Sklodowska-Curie Fellowship Programme. We are very fortunate to have passionate colleagues working to expand access to the field of advanced reactors and sharing the ideas for the future of nuclear power.

As we get ready to look back on this event, please be reminded on the FR 22 Conference website, where you will be able to find all submitted papers and working material.

In conclusion, some final thanks. Starting with our scientific secretaries, Ms. A. Gonzalez Espartero and Mr. V. Kriventsev, along with their associates Mr. Joseph Mahanes, Ms. Nikoleta Morelová, Ms. Alina Constantin and Kzrysztof Otlik, as well as Mr. Sanjai Padmanabhan and Mr. Erik Paniagua-Miranda, all of them have worked hard to make this a very successful conference.

I wish you a pleasant and safe travel to your home countries and wish you all the best in your activities. I declare this International Conference on Fast Reactors and Related Fuel Cycles adjourned.

ANNEX I. CONFERENCE STATISTICAL DATA

I-1. GENERAL INFORMATION

Organized by the:	IAEA (NENP/NEFW)
Hosted by the:	IAEA (NENP/NEFW)
Location:	IAEA HQ, Vienna, Austria
Total no. of participants and observers	473
No. of participants from Member States	442
No. of participants from the developed countries	0
No. of participants from developing countries	0
No. of participants from organizations	3
No. of countries	53
No. of organizations represented (including IAEA)	4
No. of presentations	287 technical papers
	2 opening statements
	11 plenary session presentations
	3 panel presentations
	3 YGE presentations
	2 closing statements
	198 oral technical presentations
No. of posters	63 poster presentations
Scientific Secretaries	Amparo González-Espartero, NEFW
	Vladimir Kriventsev, NENP
Scientific Support	Joseph Mahanes, NENP
	Nikoleta Morelová, NENP
	Krzysztof Otlik, NENP
	Alina Constantin, NENP
Conference Coordinators	Sanjai Padmanabhan
	Erik Paniagua-Miranda

I-2. PARTICIPANTS DATA

Total Participants and Observers: 473

Participants from Member States: 442 from 52 MS

Afghanistan	2	Japan	21	Slovakia	4
Austria	1	Jordan	2	South Africa	3
Belarus	2	Kenya	2	Spain	1
Belgium	4	Korea, Republic of	12	Sweden	4
Brazil	2	Kyrgyzstan	1	Switzerland	6
Canada	1	Libya	3	Tajikistan	2
Central African Republic	1	Mauritania	1	Thailand	3
China	24	Mexico	2	Türkiye	2
Egypt	2	Mongolia	4	United Kingdom	6
France	41	Netherlands	3	United Republic of Tanzania	1
Germany	9	Nicaragua	2	United States of America	55
Ghana	1	Niger	1	Uzbekistan	2
Hungary	4	Nigeria	2	Viet Nam	2
India	13	Pakistan	1	Yemen	3
Indonesia	1	Romania	8	Zimbabwe	1
Iran, Islamic Republic of	2	Russian Federation	153		
Iraq	1	Saudi Arabia	4		
Italy	12	Singapore	1		

Participants from Permanent Observer States: 1

Palestine	1

Participants from Organizations: 3 from 2 Organizations

OECD Nuclear Energy Agency (NEA)	2
European Commission	1

ANNEX II. LIST OF PAPERS

All materials related to oral contributions — including peer-reviewed papers, submitted presentations, and videos — are available on the FR22 Conference website:
https://conferences.iaea.org/e/FR22

Track	Paper	Presenter	Country	Paper Title
1	006	T. Beck	France	Sketch Design of Fuel Sub-Assemblies for a SFR-150 MWe
2	007	H. Yamano	Japan	Activities of the GIF Safety and Operation Project of Sodium-Cooled Fast Reactor Systems
3	008	S. Hirooka	Japan	Recent studies on fuel properties and irradiation behaviors of Am/Np-bearing MOX
3	010	T. Segawa	Japan	Development of simplified fuel fabrication technologies for fast reactors
3	011	M. Aziz Ibrahim	Egypt	Analysis of Fuel Burnup and Safety Parameters of Gas Cooled Fast Breeder Reactors
5	012	N. Krauter	Germany	Coolant flow monitoring with an Eddy Current Flow Meter at a mock-up of a liquid metal cooled fast reactor
6	014	A. Moisseytsev	United States of America	Simulation of FFTF Individual Reactivity Feedback Tests with SAS4A/SASSYS-1 Code
2	018	Y. Onoda	Japan	Modelling and Simulation of Source Term for Sodium-Cooled Fast Reactor under Hypothetical Severe Accident: Primary System/Containment System Interface Source Term Estimation
5	019	K. Aizawa	Japan	Investigation on natural circulation for decay heat removal in reactor vessel of sodium-cooled fast reactor
7	020	A. Chebeskov	Russian Federation	Specific Features of the Export of Russian Technologies of Fast Reactors and A Closed Nuclear Fuel Cycle
6	021	A. Uchibori	Japan	Development of Integrated Severe Accident Analysis Code, SPECTRA for Sodium-cooled Fast Reactor
2	022	K. Kamiyama	Japan	A Status of Experimental Program to Achieve In-Vessel Retention during Core Disruptive Accidents of Sodium-Cooled Fast Reactors

Track	Paper	Presenter	Country	Paper Title
6	024	N. Doda	Japan	Development of Multi-level Simulation System for Core Thermal-hydraulics Coupled with Plant Dynamics Analysis - Prediction of Transient Temperature Distribution in a Subassembly under Inter-subassembly Heat Transfer Effect -
8	025	K. Legkikh	Russian Federation	Experience of Operational Chemical Cleaning of BN-600 Steam Generator Evaporators rrom Corrosion Product Deposits
7	026	A. Yegorov	Russian Federation	Comparative multi-criteria analysis of scenarios of the Russian nuclear energy development in the context of uncertainty knowledge about the future
7	027	A. Yegorov	Russian Federation	Modeling the optimal economic structure of a global deploying nuclear power system with fast and thermal reactors in a partially closed nuclear fuel cycle
3	028	A. Gulevich	Russian Federation	The initial stage of closing the NFC of two-component nuclear power. Challenges and solutions
8	030	V. Smykov	Russian Federation	Problems of Decomissioning Fast Reactors And Ways of Their Solution on the Basis of The BR-10 Research Reactor
6	034	J. Dietz	Switzerland	MSR Fuel Cycle and Thermo-Dynamics Simulations
6	036	J. Krepel	Switzerland	Spatial interdependence of safety related effects in ESFR-SMART core
4	039	A. Alomari	Saudi Arabia	Creep and Creep-Fatigue Behavior of an Advanced Stainless Steel (Alloy 709) - Application to Sodium-Cooled Fast Reactors
4	041	V. Alekseev	Russian Federation	Investigation of sodium purification
3	045	E. Rodina	Russian Federation	Fuel cycle closure for high power fast neutron reactor
6	046	A. Jiménez-Carrascosa	Switzerland	Development of an Artificial Neural Network for predicting spatial interdependencies of reactivity effects in Sodium Fast Reactors
3	047	Z. Hózer	Hungary	Selection, testing and development of qualification procedure for ALLEGRO gas-cooled fast reactor fuel

Track	Paper	Presenter	Country	Paper Title
6	050	M. Messner	United States of America	A statistical design method for steady state creep applied to Grade 91 components
6	051	C. Fiorina	Switzerland	Simulation of fission gas release in the 3-D fuel performance code OFFBEAT
6	052	R. Lopez-Solis	Mexico	Verification of the SPL module of the neutron diffusion code AZNHEX through Neutronics Benchmark of CEFR Start-Up Tests
4	056	A. Orlov	Russian Federation	State of Development of Lead Coolant Technology Components for BREST-OD-300 Reactor
2	059	I. Pakhomov	Russian Federation	The Severe Accident Management of the high-power SFR with loss of the heat removal from the core
3	061	A. Gulevich	Russian Federation	Feasibility Study of Heterogeneous Transmutation of Americium in Fast Reactors
6	062	F. Roelofs	Netherlands	Dutch Thermal Hydraulic Design and Safety Support for LMFRs
3	063	A. Shadrin	Russian Federation	Fabrication and reprocessing of mixed uranium-plutonium nitride fuel for reactor BREST
6	067	X. Jia	China	Verification of SARAX Code for the Transient Analysis of Sodium-cooled Fast Reactor
2	068	P. Rajendrabhai Patel	India	Mechanistic Modelling of Aerosol Evolution in an SFR Containment Following a Hypothetical Severe Accident
2	070	X. Jin	China	Safety Analysis of Small Modular Sodium Fast Reactors in Anticipated Transients Without Scram Scenarios
7	071	N. Salnikova	Russian Federation	Export Potential and Commercialization Conditions of Fast Reactors Considering Non-Proliferation Items
6	079	N. Mosunova	Russian Federation	Codes of new generation – sustainable platform for numerical modeling of installations in the Proryv project
4	080	S. Mingyue	China	A novel method of manufacturing a heavy integrated support ring in fast reactor
6	083	J. Genin	France	First fully adjusted set of parameters for the corrosion product contamination code OSCAR-Na

Track	Paper	Presenter	Country	Paper Title
6	084	V. Chudanov	Russian Federation	Current status of development of 3D DNS CONV-3D code: one- and two-phase flow models
6	085	E. Usov	Russian Federation	Models of the integral EUCLID/V2 code for numerical simulation of severe accidents in a sodium-cooled fast reactor with MOX and MNUP fuels
5	086	A. Quaini	France	France-Japan Collaboration on Thermodynamic and Kinetic Studies of Core Material Mixture in Severe Accidents of Sodium-Cooled Fast Reactors
1	087	M. Caramello	Italy	The Status of the ALFRED Project
6	088	A. Fedorovskii	Russian Federation	Digital Technologies for Project Development ODEC and PEC and Digital Twins
5	089	D. Klinov	Russian Federation	Experimental capabilities of the research reactor facility MBIR. Main areas of the research programme in the interests of the Generation 4 reactors
6	091	A. Palagin	Russian Federation	Progress in system thermohydraulic code HYDRA-IBRAE/LM models development for fast reactor simulation
1	092	A. Alemberti	Italy	Status of Generation-IV Lead Fast Reactor Activities
6	093	R. Chalyy	Russian Federation	SOCRAT-BN Integral Code: Development, Validation and Current Status
6	094	A. Zadorozhnyi	Russian Federation	Mechanistic code BERKUT-U: self-consistent modeling of fuel rods thermomechanical behavior and processes in the fuel of fast breeder reactors
6	095	A. Sorokin	Russian Federation	Aerosol module for modeling of the fission product behavior in FR cooling circuits and NPP compartments
6	097	D. Veprev	Russian Federation	Models of the integral EUCLID/V2 code for numerical modeling of different regimes of lead-cooled fast reactor
6	104	X. Huo	China	CEFR Physical Start-Up Tests: the Core Specifications and Experiments
4	106	K. Toyota	Japan	Material Data Acquisition Activities to Develop the Material Strength Standard for Sodium-cooled Fast Reactors

Track	Paper	Presenter	Country	Paper Title
6	107	N. Pribaturin	Russian Federation	Experimental modeling of a fuel element simulator vibration in a coolant flow
5	112	D. Gugiu	Romania	Implementation of LFR Experimental Infrastructures in Romania
6	116	B. Forno	France	Phénix Control Rod Withdrawal test analysis using a multiphysics methodology
3	119	A. Tuzov	Russian Federation	Heterogeneous Burning of Minor Actinides in A Fast Reactor
2	123	A. Pantano	France	PRE-DESIGN OF A PASSIVE DECAY HEAT REMOVAL SYSTEM WITH A PHASE CHANGE MATERIAL FOR SMR-SFR
3	124	M. Krivov	Russian Federation	Uranium and mixed uranium-plutonium nitrides thermal stability
2	126	A. Bachrata	France	Development of methodology to evaluate mechanical consequences of vapor expansion in SFR severe accident transients: lessons learned from previous France-Japan collaboration and future objectives and milestones
1	127	B. Merk	United Kingdom of Great Britain and Northern Ireland	iMAGINE - a Breakthrough Technology for Closing the Fuel Cycle without Reprocessing
7	128	D. Settimo	France	TECHNICAL AND ECONOMICAL FEATURES OF COMMERCIAL SODIUM FAST REACTOR IN FRANCE
6	129	A. Ushatikov	Russian Federation	Application of Digital Twin of Fast Reactor Plant for Control System Algorithm Testing
8	131	A. Izhutov	Russian Federation	BOR-60 Reactor Operating Experience, Work on Improving Safety and Extending Lifetime
6	132	M. Szieberth	Hungary	Comparison of calculation methods for lead cooled fast reactor reactivity effects
3	134	S. Kviatkovskii	Russian Federation	Multi-criteria comparison of the efficiency of minor actinides burning in different nuclear reactors based on the INPRO/IAEA KIND approach

Track	Paper	Presenter	Country	Paper Title
1	136	S. Shepelev	Russian Federation	Development of BN Reactor Technology in Russia
3	138	M. Edmondson	United Kingdom of Great Britain and Northern Ireland	The role of pyrochemical processing in a NetZero economy in the UK
8	141	D. Villani	France	Treatment of sodium of Superphenix Fast Breeder Reactor
7	142	I. Zhuravlev	Russian Federation	Effect of Reactor Technology on Economics of SMR Projects
3	143	A. Gulevich	Russian Federation	On the Possibility to Change the Isotopic Composition of Plutonium from the Spent MOX Fuel of PWRS in Fast Reactors
7	146	E. Marova	Russian Federation	Effective Fuel Supply of Two-Component Nuclear Energy System with VVER-BN Reactors
6	147	L. Mesthiviers	France	Study on actinide conversion capabilities of Molten Salt Reactors (MSR)
5	148	I. Kuzina	Russian Federation	Thermohydraulic Tests in Justification of Design Characteristics of the BREST-OD-300 RP Steam Generator
3	150	C. Jensen	United States of America	Advanced Reactor Experiments for Sodium Fast Reactor Fuels (ARES) Project: Transient Irradiation Experiments for Metallic and MOX Fuels
6	151	Y. Liu	China	The fluid structure interaction of narrow gaps between thin-wall coaxial structures in fast reactors
2	152	T. Le Meute	France	Modelling of postulated reactivity insertion in a Generation IV Molten Salt Reactor
2	155	F. Payot	France	The SAIGA in-pile experimental program to qualify the SIMMER calculation tool in SFR Severe Accident Conditions
3	156	Y. Kotov	Russian Federation	Potential Role of Fast Reactors with Heterogeneous Fuel Assembly in Development Nuclear Power Structure

Track	Paper	Presenter	Country	Paper Title
6	163	A. Gomez Torres	Mexico	Verification and validation of neutronic codes using the start-up fuel load and criticality tests performed in the China Experimental Fast Reactor
3	164	L. Capriotti	United States of America	Postirradiation characterization of AFC metallic fuel alloys concepts.
6	166	I. Bukhtiiarov	Russian Federation	The solution of nuclide kinetic equation for fast reactor in the OpenBPS code with options of choosing calculation method and uncertainties analysis.
9	167	G. Tikhomirov	Russian Federation	Training of New Generation Specialists in the Field of Fast Neutron Reactors and Nuclear Fuel Cycle Closure
4	175	S. Liu	China	Fabrication and Performance Assessment of ODS FECRAL Cladding Tube
4	178	S. Wei	China	The δ-ferrite transformation behavior and mechanical properties of 316H weld metal during high temperature service
5	187	J. Kuzina	Russian Federation	Experimental and computational studies of heat exchange for liquid metals boiling in fuel assembly models at accidental conditions
1	190	S. Rukhlin	Russian Federation	Optimization of Built-In Primary Sodium Purification System for Advanced BN Reactor Plant
2	194	M. Konstantin	Switzerland	Coupled neutronic/thermal-hydraulic simulation of Unprotected Loss of Flow Test at Fast Flux Test Facility
5	197	M. Grushko	Russian Federation	Experimental test facility to test a prototype of the air heat exchanger gate for the advanced bn reactor plant. Design and construction items
6	199	B. Kvizda	Slovakia	Recent thermal hydraulic studies of Gas Fast Reactor demonstrator ALLEGRO
6	200	D. Uralov	Russian Federation	Possibility of Simulating Natural Circulation in Fast Neutron Reactors Using a Light Water Test Facility
9	201	A. Nakhabov	Russian Federation	Preserving and transferring knowledge in the field of fast reactor technologies. Experience of the Obninsk Institute of Nuclear Power Engineering MEPhI

Track	Paper	Presenter	Country	Paper Title
2	204	J. Droin	France	Integrating safety at the first design stages: a new methodology for safety-oriented SFR core design
3	205	Y. Karazhelevskaia	Russian Federation	The influence of isotopic composition of plutonium in fast reactor fuel on the reactivity margin
5	206	J. Kuzina	Russian Federation	Physical modeling of hydrodynamics and heat exchange in fast reactors with liquid metal coolants
1	207	J. Guidez	France	Optimization of the secondary loops on the ESFR SMART project
6	208	S. Belov	Russian Federation	APPROACHES TO FORM THE BN 1200 CORE START LOADING USING MOX-FUEL AND MNUP-FUEL
6	209	S. Belov	Russian Federation	DISTINCTIVE FEATURES OF THE BN-800 CORE IN THE COURSE OF TRANSITION TO COMPLETE MOX-FUEL LOADING
2	210	J. Guidez	France	Application of the practical elimination concept within the framework of the ESFR-SMART project to improve the intrinsic safety of the sodium-cooled fast reactor
3	213	A. Belyaeva	Russian Federation	RESULTS OF POST-IRRADIATIONS EXAMINATIONS OF MIXED NITRIDE PINS WITH GAS AND LIQUID METAL SUB-LAYERS
8	217	M. Thangamani	India	Operating Experience of FBTR
1	219	S. Dmitrii	Russian Federation	Project of a multipurpose lead reactor with a hard neutron spectrum
7	220	P. Li	China	Development status of commercial SMRs and its experience to China
5	224	I. Kuzina	Russian Federation	Complex of experimental facilities for design and safety justification of fast reactors with liquid metal coolants
3	232	P. Gantsovsky	Russian Federation	RADIATION AND HYGIENE ASSESSMENT OF EXTERNAL EXPOSURE FACTORS OF PERSONNEL WORKING AT EXPERIMENTAL FACILITIES IN THE PRODUCTION OF MIXED NITRIDE URANIUM-PLUTONIUM FUEL
6	233	T. Kyum Kim	United States of America	Neutronics Benchmark of CEFR Start-Up Tests: Reaction Rates and Reactivity Coefficients

Track	Paper	Presenter	Country	Paper Title
5	238	M. Weathered	United States of America	Overview of a Sodium Fast Reactor Thermal Hydraulic Test Facility
3	239	A. Komarov	Russian Federation	TYPES OF CHEMICAL COMPOUNDS IN THE ASSESSMENT OF RADIATION AND HYGIENIC HAZARDS WHEN WORKING WITH IRRADIATED NITRIDE FUEL
2	240	T. Sumner	United States of America	Overview of the Versatile Test Reactor Safety Analysis
2	241	T. Sumner	United States of America	Safety Analysis of the ARC-100 Sodium-Cooled Fast Reactor
4	252	E. Kent	United States of America	Gear Test Assembly: First Liquid Metal Component Testing in METL
4	254	M. Orlov	Russian Federation	Thermally conductive liquid-metal sublayer in fuel element
2	258	J. Andrus	United States of America	Development of the Versatile Test Reactor (VTR) Probabilistic Risk Assessment
3	262	Y. Khomyakov	Russian Federation	Physical feasibility of MA transmutation in a two-component nuclear energy system in Russia
2	268	J. Andrus	United States of America	Application of a Risk-Informed Performance-Based Approach for the Authorization of the Versatile Test Reactor
1	269	A. Kato	Japan	Progress in conceptual design of a pool-type sodium-cooled fast reactor in Japan
2	270	S. Kubo	Japan	France-Japan Collaboration on the SFR Severe Accident Studies: Outcomes and future work program
1	271	S. Jang	Korea, Republic of	Conceptual design of ultra-long life hybrid micro modular reactor cooled by potassium heat pipe
6	272	F. Bostelmann	United States of America	Objectives and Status of Neutronics Sub-exercises of the OECD/NEA Benchmark for Uncertainty Analysis in Modelling for Design, Operation and Safety Analysis of SFRs
1	275	T. Dong Cao Nguyen	Korea, Republic of	Core Design of 100MWe Advanced Nitride-fueled Simplified Liquid Metal Cooled Fast Reactor

Track	Paper	Presenter	Country	Paper Title
2	277	J. E. Chang	United States of America	Modeling and Simulation of Source Term for Sodium-Cooled Fast Reactor under Hypothetical Severe Accident: Sodium Fire and Radionuclide Transport in Containment
6	281	J. Choe	Korea, Republic of	Neutronics Benchmark of CEFR Start-Up Tests: Temperature Coefficient, Sodium Void Worth, and Swap Reactivity
6	285	N. Morelova	IAEA	Neutronics Benchmark of CEFR Start-Up Tests: An IAEA coordinated research project
3	300	M. Khramtsov	Russian Federation	Research and development of nuclear fuel for fast neutron reactor
2	302	C. Journeau	France	French-Japanese experimental collaboration on fuel-coolant interactions in sodium-cooled fast reactors
1	304	P. Vácha	Czech Republic	GFR Research and Development Programme in V4 countries
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6	316	A. Bachchan	India	Neutronics analysis of CEFR Start-up tests at IGCAR using FARCOB and ERANOS 2.1 Code Systems
4	317	P. Reddy G. V.	India	Creep and Tensile Properties of Indian Advanced Fast Reactor Clad tubes (IFAC-1) for Future FBRs
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3	319	A. Bakhin	Belarus	LOW ENRICHMENT NUCLEAR FUEL BASED ON URANIUM-ZIRCONIUM CARBONITRIDE: REACTOR TESTS AND PREPARATION FOR STUDIES AT CRITICAL ASSEMBLIES
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4	332	R. Mythili	India	Development of Plasma Nitriding as alternate hardfacing technique for Large components of FBR and Assessment of static In-Sodium Stability of Plasma Nitrided Layer
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2	338	K. Tucek	European Commision	System Safety Assessment of the Generation IV Lead Fast Reactor
5	340	C. Grandy	United States of America	Mechanisms Engineering Test Loop (METL) Facility
1	341	D. Gérardin	France	Evaluation of an increase of the power density for the French commercial Sodium Fast Reactor and optimization study at 1100 MWe with the SDDS tool
1	343	M. Caramello	Italy	Integration of Small Modular Lead Fast Reactor with Energy Storage for load-following operation in high V-RES penetration electricity markets
5	345	G. Firpo	Italy	ALFRED High priority R&D Needs
6	350	W. Klein-Heßling	Germany	Regulatory Perspectives on Analytical Codes and Methods for Advanced Reactors
2	351	M. Bucknor	United States of America	The Versatile Test Reactor (VTR) Approach to Sodium Fire Hazards Analysis and Protection System Methodology
5	357	A. Zagornov	Russian Federation	MULTIPURPOSE RESEARCH FACILITY MBIR AND POLY FUNCTIONAL RADIOCHEMICAL COMPLEX (R&D COMPLEX) AS A UNIQUE RESEARCH PLATFORM
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6	370	O. Bovati	United States of America	CFD Simulations on a hexagonal 61-pin wire-wrapped fuel bundle with STARCCM+ and comparison with experimental data.
2	376	I. Shvetsov	Russian Federation	ANALYSIS OF THE SGTR ACCIDENT FOR SAFETY JUSTIFICATION OF TWO-CIRCUIT LEAD COOLED REACTOR.
5	379	Y. Sokolov	Russian Federation	Overview of critical experiments with fast metal cores held on assembly machine FKBN-2
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7	385	O. Komlev	Russian Federation	TECHNOLOGICAL SUPPORT OF THE NON-PROLIFERATION FOR SVBR-100 FUEL CYCLES
1	387	G. Toshinskii	Russian Federation	CHOICE OF A COOLANT FOR A MODULAR SMALL POWER REACTOR SVBR-100
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6	422	D. Maurizio Castelluccio	Italy	REALISATION OF AN ADJUSTED NUCLEAR DATA LIBRARY BASED ON ENDF/B-VIII.0 NUCLEAR DATA EVALUATIONS FOR THE ALFRED CORE
5	423	K. Weaver	United States of America	Versatile Test Reactor (VTR) Experimental Capabilities
9	424	M. Apostol	Romania	Investigation on Human Resources Needs and Competences Building for ALFRED Implementation in Romania
1	432	R. Hill	United States of America	Overview of U.S. Fast Reactor Technology R&D Program
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5	435	J. Such	France	Overview of the R&D programs led by the past at IRSN on sodium fire
5	436	J. Roglans-Ribas	United States of America	Versatile Test Reactor (VTR) Project Mission and Status
9	437	P. Paviet	United States of America	GEN IV INTERNATIONAL FORUM WEBINARS INITITATIVE
6	440	E. Ivanov	France	Target Accuracy Requirements and an evidence-based background for MSFR safety assessment
2	449	S. Gianfelici	Italy	Transient 3D simulations for the ASTRID reactor: preliminary results for the ULOF initiation phase
6	459	R. Kumar Maity	India	Computational fluid dynamics study for estimation of dilution for failed fuel location system
2	461	A. Samantara	India	Thermal hydraulic assessment of the performance of secondary sodium system based decay heat removal circuit
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5	476	S. Pathak	India	AN EXPERIMENTAL STUDY ON SECONDARY SODIUM SYSTEM BASED DECAY HEAT REMOVAL CIRCUIT OF A SODIUM COOLED FAST REACTOR
1	477	J. Mote	India	DESIGN & ANALYSIS OF A NOVEL ARRANGEMENT FOR COUPLING AND DECOUPLING OF ROTATABLE PLUGS IN PFBR
1	478	S. Raghupathy	India	Progress in the Design and R&D for future FBRs
2	488	S. Raghupathy	India	Design Studies Towards Raising FBTR to Full Power
8	492	V. Padmanabhan	India	Design, Experimental trials and Qualification of explosive welding technique for plugging of degraded PFBR Steam Generator tubes
3	493	D. Narasimhan	India	Advanced Flow-Sheet for Partitioning of Trivalent Actinides from Fast Reactor High Active Waste
8	494	J. Jose	India	Advanced in-situ Calibration and Probe Release Mechanism for PFBR SG Inspection System (PSGIS)
8	500	C. Ramalingam	India	Reactor Core Viewing System for the pre-commissioning stage inspection of reactor core components of Prototype Fast Breeder Reactor
1	502	P. Kumar Patel	India	Design of secondary sodium based decay heat removal system for future fast breeder reactors

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3	504	N. Sivayya Dudala	India	Root Cause Analysis of FBTR Failed Fuel Pin
2	509	S. Rajagopalan	India	Over three decades of radiological protection experience at Fast Breeder Test Reactor (FBTR)
4	510	K. Tucek	European Commision	Tensile testing of sub-sized T91 and 316L steel specimens in liquid lead
5	511	F. Heidet	United States of America	VERSATILE TEST REACTOR: CONCEPTUAL CORE DESIGN OVERVIEW
8	513	A. Jyothishkumar	India	Experience in Preheating of PFBR Reactor Assembly
8	514	N. Sahu	India	Commissioning and Operating Experience for Secondary Sodium Systems and is Auxiliaries of PFBR
9	516	N. Barron	United Kingdom of Great Britain and Northern Ireland	An e-learning tool on Fast Reactors and their Fuel Cycles
6	521	D. Wise	United States of America	Passive Heat Removal System Analysis for the Westinghouse Lead Fast Reactor
1	528	P. Ferroni	United States of America	The Westinghouse Lead Fast Reactor: overview and progress in development
6	534	A. Moisseytsev	United States of America	BLIND PHASE RESULTS FOR TRANSIENT SIMULATIONS OF THE FFTF LOSS OF FLOW WITHOUT SCRAM TEST #13
6	536	N. Stauff	United States of America	Blind-Phase Results of the FFTF Neutronic Benchmark
4	537	C. Latge	France	Sodium coolant: interaction with its environment and coolant processing
2	538	P. Calle Vives	IAEA	EXAMPLES OF AREAS OF NOVELTY IN LIQUID METAL FAST REACTORS TO CONSIDER IN THE REVIEW OF APPLICABILITY OF THE IAEA SAFETY STANDARDS: FISSION PRODUCT RETENTION BARRIERS: DIFFERENCES BETWEEN LIQUID

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3	544	A. Dedyukhin	Russian Federation	Investigation of the anodic processes on the ceramic anode in the oxide-chloride melts
7	550	A. Bychkov	IAEA	The INPRO project studies on the double-component nuclear power systems with the closed fuel cycle and fast reactors: past and future
	552	H. Yang	China	China Key Note
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	556	L. Chae Young	Republic of Korea	Republic of Korea Key Note
	557	V. Pershukov	Russian Federation	Russian Federation Key Note
	558	A. Caponiti	United States	United States Key Note
	559	M. Betti	European Commission	European Commission (EC) Key Note
	560	B. Hill	Generation IV International Forum	Generation IV International Forum Key Note
	561	T. Ivanova	OECD/Nuclear Energy Agency	OECD/Nuclear Energy Agency Key Note
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	564	R. Grossi	IAEA	IAEA Opening Address
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	566	K. Kumar Pal	India	YGE Winner: Advanced Functional Materials for Next-Generation Fuel Reprocessing
	567	O. Kucheryavykh	Russian Federation	YGE Winner: Production of Mo-99 isotope in the BN reactor by beryllium blocks
	568	T. Zhe Chuan	Singapore	YGE Winner: Small Modular Fast Reactors for the ASEAN Region: Implementation of the TRISO Fuel Particle Concept as a Regional Variant of the Fast Reactor
	569	A. Kumar Bhaduri	India	Chair of Conference International Advisory Committee Closing Remarks
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ANNEX III. LIST OF POSTERS

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1	108	R. Pineda	China	Simple Design Comparison of uranium nitride pin cell assembly and matrix fuel assembly for a Lithium Cooled Fast Reactor
1	122	I. Drobyshev	Russian Federation	Hybrid high power fast breeder reactor with metallic fuel and additives consisting with lightweight atoms
1	158	E. Kulikov	Russian Federation	On substantial slowing down of the kinetics of fast transient processes in fast reactor
1	159	E. Kulikov	Russian Federation	Investigation of characteristics of fast power reactor with an additional function of large-scale production of plutonium-238
1	286	T. Kim	Korea, Republic of	Leak-Before-Break Design of Double-Walled Once-Through Steam Generators for Lead Cooled Fast Reactor
1	311	J. Bousquet	Germany	New Finite Element Neutron Kinetics Code System FENNECS/ATHLET for Coupled Safety Assessment of (Very) Small and Micro Reactors
1	365	S. Fomin	Ukraine	POWER CONTROL OF THE FAST NUCLEAR-BURNING-WAVE REACTOR
1	378	A. Petrenko	Russian Federation	INDUSTRIAL ENERGY COMPLEX WITH FAST NEUTRON REACTOR
1	398	K. Hoang Van	Viet Nam	Development of Burnup Analysis System for rotational and Spiral Fuel Shuffling scheme in Breed-and-Burn Fast Rectors

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1	414	D. Visser	Netherlands	CFD ANALYSES OF THE ALFRED HOT PLENUM
1	434	F. Lodi	Italy	The "ALFRED White Book": a business card of the project
1	501	S. Aithal	India	DESIGN & DEVELOPMENT OF CUSTOM SHAPED BACK-UP SEAL IN SILICONE FOR PFBR
1	551	R. El-Emam	Canada	LARGE-SCALE HYDROGEN PRODUCTION; Fast-neutron Reactors Coupled to Thermochemical Copper-Chlorine Hydrogen Plant
2	054	T. Ishizu	Japan	Model validation of the ASTERIA-SFR code related to freezing phenomena of liquid and liquid/particle mixtures based on THEFIS experimental results
2	069	P. Rajendrabhai Patel	India	Modelling of radionuclide release from primary system during a hypothetical severe accident in an SFR
2	174	J. Wang	China	Study on Sodium Fire PSA Methodology for Pool-Type Sodium cooled Fast Reactor
2	180	H. Sonoda	Japan	Development of in-vessel source term evaluation method for ULOF events in sodium-cooled fast reactor
2	203	Y. Du	China	Experimental and Numerical Study on Temperature Fluctuation in The Upper Plenum of Fast Reactor
2	248	J. Yang	China	Analysis of sodium fire accident after upgrade of ventilation system of primary loop's corridor
2	259	D. Grabaskas	United States of America	Development of the Simplified Radionuclide Transport (SRT) Code Version 2.0 for Versatile Test Reactor (VTR) Mechanistic Source Term Calculations
2	260	E. Orlova	Russian Federation	Increase of nuclear power plant hydrogen safety using zirconium accumulator

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2	368	B. Hollrah	United States of America	SIMULATION OF THE FAST FLUX TEST FACILITY LOSS-OF-FLOW WITHOUT SCRAM ACCIDENT SCENARIO USING THE SAM COMPUTER CODE
2	391	S. Shahbazi	United States of America	Fast Reactor Source Term Modeling and Simulation Functional Requirements and Gap Assessment
2	462	V. Govindarajan	India	Thermal Hydraulic Simulation of Loss of Flow Without Scram Test in FFTF using DYANA-P code
3	009	M. Nishina	Japan	Development of density control technologies for MOX pellet using dry recycled powders
3	064	V. Vidanov	Russian Federation	R&D on recovery and separation of americium and curium under "Proryv" project
3	157	E. Kulikov	Russian Federation	Controlled thermonuclear fusion: potential role of a joint (Th-U-Pu) nuclear fuel cycle
3	161	O. Ashraf	Russian Federation	Transmutation efficiency of minor actinides in fast-and thermal-spectrum molten salt reactors
3	162	A. Savelev	Russian Federation	Revealing the dependencies of partitioning americium-241 and uranium using sorption technology based on solid-phase extractant TODGA
3	168	A. Terekhova	Russian Federation	Transmutation of minor actinides in a fast reactor with uranium-curium fuel
3	231	A. Samoylov	Russian Federation	COMPLEX RADIATION AND HYGIENE STUDIES OF RADIATION IMPACT FACTORS ON PRODUCTION PERSONNEL, MIXED NITRIDE URANIUM-PLUTONIUM FUEL FOR FAST NEUTRON REACTORS

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3	384	P. Sannikova	Russian Federation	INVESTIGATION OF THE SOLUBILITY OF ACTINIDE FLUORIDES FOR THE CHOICE OF A SALT SOLVENT FOR A MOLTEN-SALT REACTOR-BURNER OF MINOR ACTINIDES
3	428	A. Kumari	India	Removal of Radiocesium from High-Level Liquid Waste using Inorganic Ion-exchangers
3	481	A. Krishnamurthy	India	Optimization of Ruthenium concentration in PUREX Process during Fast reactor fuel Reprocessing
3	484	P. Sivakumar	India	Assay of Waste drum based on Passive Neutron Counting Technique
3	490	K. Dhananjeya	India	Design, manufacturing and transportation of high capacity High Level Liquid Waste Storage tanks
3	491	A. Krishnamurthy	India	Evaluation of EPDM and Silicone rubber compounds for application in Reprocessing Plant
3	495	A. Krishnamurthy	India	Development of Artificial Intelligence through PLC & SCADA to predict process related failure and abnormality in a Reprocessing Plant
3	541	A. Potapov	Russian Federation	Reprocessing of nitride and metallic spent nuclear fuel using molten salts
3	543	A. Potapov	Russian Federation	Electrical conductivity of multicomponent chloride melts, containing ions of mono-, di-, and trivalent metals
3	545	Y. Zaikov	Russian Federation	Determination of the metallic and oxide compounds in models based on metallic uranium containing uranium dioxide, metallic neodymium, cerium as well as neodymium and cerium oxides

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4	055	D. Obukhov	Russian Federation	DEVELOPMENT OF SUBMERGED ELECTROMAGNETIC PUMP FOR LIQUID LEAD
4	057	R. Askhadullin	Russian Federation	CURRENT STATE AND ISSUES OF THE HEAVY LIQUID METAL COOLANT TECHNOLOGY DEVELOPMENT (PB, PB-BI)
4	066	E. Kinev	Russian Federation	The working capacity analysis of boron carbide after two-year operation as an emergency protection material of the fast reactor
4	082	N. Loginov	Russian Federation	ON MEASUREMENT OF OXYGEN CONCENTRATION IN SODIUM BY MEANS OF PLUG INDICATOR
4	102	Q. Chen	China	Irradiation-Thermo-Mechanical Coupling Analysis and Calculation of Fast Neutron Oxide Fuel Element
4	236	D. Wu	China	Influence of preheating temperature on delta-ferrite formation and mechanical properties of 12%Cr steel weld metals
4	264	O. Golosov	Russian Federation	NON-DESTRUCTIVE METHOD FOR DETERMINING STEEL CORROSION COEFFICIENTS IN LEAD
4	298	V. Federiaeva	Russian Federation	HEAT TRANSFER CALCULATION AND SERVICE LIFE TIME ESTIMATION OF SUBMERGED ELECTROMAGNETIC PUMP FOR LIQUID LEAD
4	374	J. Li	China	Irradiation Effects of T91 Ferritic/Martensitic Steel
4	393	O. Sambuu	Mongolia	Impact of Core Materials on The Fuel cladding Irradiation Damage in Breed-and-Burn Fast Reactors

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4	397	T. Obara	Japan	Impact of Cladding Material on Neutronic Balance in Breed-and-Burn fast reactors
4	445	C. Avinash	India	Experimental study on sodium insulation interaction and its effect on structural material
5	013	G. Gerbeth	Germany	Eddy Current Flow Meter flow rate measurements in liquid Sodium at the SUPERFENNEC loop
5	235	A. Karev	Russian Federation	RADIATION-HYGIENIC ASSESSMENT OF INTERNAL EXPOSURE FACTORS OF PERSONNEL WORKING AT EXPERIMENTAL FACILITIES IN THE PRODUCTION OF MIXED NITRIDE URANIUM-PLUTONIUM FUEL
5	244	D. Duan	China	Experimental investigation of the fluid-structure interaction effect between adjacent equipment supports in a fast reactor
5	255	M. Trapeznikov	Russian Federation	JUSTIFICATION OF CRITICAL EXPERIMENTS ON STAND FKBN-2 TO VERIFY NEUTRON-PHYSICAL SOFTWARE FOR CALCULATIONS OF THE MOLTEN-SALT REACTOR
5	312	R. Ganapathy	India	Conceptual Core configuration for increasing Power of Fast Breeder Reactor to 40 MWt
5	348	M. Caramello	Italy	ALFRED DHR system scaling verification and numerical pre-test analysis
5	366	A. Nelson	United States of America	VERSATILE TEST REACTOR: CORE SYSTEM DESIGN REQUIREMENTS TO SUPPORT ADVANCED REACTOR DEVELOPMENT
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6	033	J. Krepel	Switzerland	Characterization of the Molten Chloride Fast Reactor fuel cycle options
6	053	R. Lopez-Solis	Mexico	Verification and validation of the CEFR Serpent model for the generation of reference solutions and Cross Sections database for the deterministic code AZNHEX
6	058	O. Miazdrikova	Russian Federation	Modeling of water leak into sodium in the BN-600 steam generator
6	096	A. Belov	Russian Federation	Calculation of the materials activation with BPSD code
6	098	V. Bereznev	Russian Federation	Integral code COMPLEX for radiation safety assessment of reactor and nuclear fuel cycle facilities
6	110	R. Shamsutdinov	Russian Federation	Experience of Using CFD Models for Development of High-Temperature Furnace Equipment for Fabrication of Mixed Nitride Uranium-Plutonium Fuel Pellets
6	135	A. Moise	Romania	Neutronic Calculation of CEFR Core using Different Nuclear Data Libraries
6	165	A. Ali	Egypt	ADS for Energy Production and 233U breeding in HEU-Thorium Oxide system
6	179	I. Fadeev	Russian Federation	ELABORATION OF THE THERMAL-HYDRAULIC CHARACTERISTICS OF THE REACTOR PLANT BASED ON THE OPERATION EXPERIENCE OF THE POWER UNIT WITH BN-800 REACTOR
6	216	S. Rogozhkin	Russian Federation	DEVELOPMENT OF THE TECHNICAL APPROACH FOR RESEARCH OF THE SODIUM COOLANT CURRENT IN THE INTEGRAL TYPE REACTOR
6	227	V. Mishin	Russian Federation	COMPARATIVE ANALYSIS OF CALCULATIONAL AND EXPERIMENTAL DIFFERENCES OF THE NEUTRON-PHYSICAL

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				CHARACTERISTICS OF THE BN-800 REACTOR
6	245	E. Seleznev	Russian Federation	Some results of using partial equations for calculations of transient processes in fast breeder reactors
6	308	G. Yujie	China	Application of Model Based System Engineering in Design of Digital Fast Reactor Nuclear Power Plant
6	313	J. Wang	China	Research on the Impact of Advanced Rule Design System on the Digitization of Reactor Building Model
6	353	I. Popov	Russian Federation	Software for Simulation of Fast Reactor Operation in a Closed Nuclear Fuel Cycle (SC RTM-2)
6	356	I. Makeeva	Russian Federation	The Code Complex for Computational Evaluation of Technical Solutions and Optimization of Processes Parameters of CNFC
6	416	R. Pergreffi	Italy	Modeling of the coolant region in the ALFRED core in case of thermal expansion
6	421	D. Maurizio Castelluccio	Italy	ENDF/B-VIII.0 NUCLEAR DATA SENSITIVITY AND UNCERTAINTY (S/U) ANALYSIS OF KEY SAFETY-RELEVANT REACTIVITY COEFFICIENTS FOR THE ALFRED CORE
6	438	J. Claudio Fiel	Brazil	CRITICALITY SENSITIVITY ANALYSIS IN RELATION TO EMPTIES OF A FAST REGENERATOR NUCLEAR REACTOR
6	451	X. Jia	China	Study on the Method of Correction of Fast Reactor Power Distribution by MCNP
6	460	R. Kumar Maity	India	Integrated thermal hydraulic analysis of Hot and Cold Pools of a liquid sodium cooled 600 MWe fast reactor

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6	542	A. Potapov	Russian Federation	Thermodynamic simulation of the oxidation processes at the reprocessing of spent nuclear fuel in the LiCl-KCl melt
7	037	V. Ivanov	Russian Federation	NEXT GENERATION NUCLEAR POWER: RADIOLOGICAL SUSTAINABILITY AND ECOLOGICAL ADVANTAGES
7	133	M. Roslaya	Russian Federation	Two-Component Energy Industry under Conditions of Closed Nuclear Fuel Cycle: Economic Benefits
7	140	S. Kviatkovskii	Russian Federation	Sustainability of nuclear and non-nuclear power generation options under Russian conditions: a comparative evaluation study
7	223	E. Metlyaev	Russian Federation	INTEGRATED RADIATION AND HYGIENIC APPROACH TO PRODUCTION SAFETY. ASSESSMENT OF THE IMPACT ON PUBLIC HEALTH
7	226	O. Parinov	Russian Federation	RISK FACTORS OF COMPLEX RADIATION AND NON-RADIATION EFFECTS ON THE HEALTH OF PERSONNEL IN ASSESSING THE IMPACT OF THE PRODUCTION OF INNOVATIVE FUEL FOR FAST REACTORS
7	381	I. Iordache	Romania	Nuclear Hydrogen and Fast Reactors
7	548	V. Kuchinov	Russian Federation	EXPORT OF RBN WITH SNCD AND NUCLEAR PROLIFERATION RISKS
8	029	V. Smykov	Russian Federation	CORROSION HYDROGEN MASS TRANSFER IN FAST REACTOR STEAM GENERATORS OF THE SODIUM-WATER TYPE
8	038	X. Hu	China	Design of experimental scheme for activation method of China demonstration fast reactor

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8	283	Q. Zhou	China	A Study on the Development of a Procedure Complexity Evaluation and Optimization for Operating Procedures of China Experimental Fast Reactor
8	375	K. Chaturvedi	India	Numerical Investigation of Cellular Convection in the Cover Gas space of Fast Breeder Test Reactor
8	411	B. Wilcox	Nigeria	New Concepts and Methodologies for the Effective Deployment of Gen IV reactors
8	498	D. Jagadishan	India	Development of a 15 kg servo manipulator for remote handling applications
8	499	P. Bhanu	India	Novel Electrical, Electronics and Instrumentation systems for Fast Reactor Fuel Reprocessing Plants
9	293	N. Maksimov	Russian Federation	Cognitive Information Retrieval Based on Ontological Model of Knowledge Representation

ABBREVIATIONS

ADS	Accelerator-Driven System
AFC	Advanced Fuel Campaign
AHX	Air Heat Exchanger
ALFRED	Advanced Lead cooled Fast Reactor European Demonstrator
AMR	Advanced Modular Reactor
ANL	Argonne National Laboratory
ANN	Artificial Neural Network
ARC	Argonne Reactor Computation
ARES	Advanced Reactor Experiments for Sodium Fast Reactor Fuel
ASEAN	Association of Southeast Asian Nations
ASME	American Society of Mechanical Engineers
ASTRID	Advanced Sodium Technological Reactor for Industrial Demonstration
ATR	Advanced Test Reactor
ATWS	Anticipated Transient Without Scram
BDBA	Beyond Design-Basis Accident
BR	Breeder Reactor
BSA	Basis for the Safety Approach
CC	Core Catcher
CDA	Core Disruptive Accident
CEA	French Alternative Energies and Atomic Energy Commission
CEFR	China Experimental Fast Reactor
CEF	Critical Experiments Facility
CFD	Computational Fluid Dynamics
CFR	China Fast Reactor
CGHM	Cover-Gas Hydrogen Meter
CHF	Critical Heat Flux
CIAE	China Institute of Atomic Energy
CiADS	China initiative Accelerator Driven System
CICT	Constant Inlet Coolant Temperature
CNFC	Closed Nuclear Fuel Cycle
CPU	Central Processing Unit
CR	Control Rod

CRDL	Control Rod Drive Line
DDG	Deputy Director General
DDM	Device Deployment Module
DG	Design Guideline
DHR	Decay Heat Removal
DHRS	Decay Heat Removal System
DHX	Direct reactor Heat Exchanger
DND	Delayed Neutron Detector
DOE	Department of Energy
DRACS	Direct Reactor Auxiliary Cooling System
DTA	Dismountable Test Assembly
DTHM	Diffusion-Type Hydrogen Meter
EBR	Experimental Breeder Reactor
EC	European Commission
ECFM	Eddy Current Flow Meter
EFPD	Effective Full Power Day
ELF	Electrical Long-running Facility
ELFR	European Lead Fast Reactor
ELTA	Extended Length Test Assembly
ENEA	Italian National Agency for new technologies, energy and sustainable economic development
ERDF	European Regional Development Funds
ESFR	European Sodium Fast Reactor
ESS	Energy Storage System
EPR	European Pressurized water Reactor
EURATOM	European Atomic Energy Community
FA	Fuel Assembly
FBR	Fast Breeder Reactors
FBTR	Fast Breeder Test Reactor
FC	Fuel Cycle
FCCI	Fuel Cladding Chemical Interaction
FCMI	Fuel Clad Mechanical Interaction
FFTF	Fast Flux Test Facility
FIMA	Fissions per initial metal atom
FNR	Fast Neutron Reactor

FPC	Fuel Performance Code
FR	Fast Reactor
FSA	Fuel Subassembly
GAIN	Gateway for Accelerated Innovation in Nuclear
GEM	Gas Expansion Module
GFR	Gas cooled Fast Reactor
GIF	Generation IV International Forum
GPN	Gas-Phase Neutralization
GPU	Graphics Processing Unit
GTA	Gear Test Assembly
H-MMR	Hybrid Micro-Modular Reactor
HEU	High-Enriched Uranium
HFR	High Flux Reactor
HLLW	High-Level Liquid Waste
HMMR	Hybrid Micro Modular Reactor
HPRL	High Priority Request List
IAEA	International Atomic Energy Agency
IFAC	Indian Advanced Fast Reactor Clad tubes
IFRES	Integrated FR nuclear Energy System
IGCAR	Indira Gandhi Centre for Atomic Research
IGR	Impulse Graphite Reactor
IHX	Intermediate Heat exchangers
INL	Idaho National Laboratory
INPRO	International Project on Innovative Reactors and Fuel Cycles
IPFM	In-Pin Fuel Motion
ISHM	In-Sodium Hydrogen Meter
JAEA	Japan Atomic Energy Agency
JSME	Japan Society of Mechanical Engineers
KAERI	Korea Atomic Energy Research Institute
KIT	Karlsruhe Institute of Technology
KNS	Kompakter Natriumsiede-Kreislauf
LBE	Lead-Bismuth Eutectic
LCOE	Levelized Cost of Electricity
LEU	Low-Enriched Uranium
LFR	Lead Cooled Fast Reactor

LHR	Linear Heat Rate
LMC	Liquid Metal Chromatography
LMFR	Liquid Metal Cooled Reactor
LMP	Larson-Miller Parameter
LOR	Lowering Of control Rods
LORL	Loss Of Reactor Level
LRP/SRP	Large and Small Rotatable Plugs
LWR	Light Water Reactor
M&S	Modelling and Simulation
MA	Minor Actinide
MBIR	Multi-Purpose Fast Neutron Research Reactor
METL	Mechanisms Engineering Test Loop facility
MNUP	Mixed Uranium-Plutonium Nitride
MoU	Memorandum of Understanding
MOX	Mixed Oxide
MSFR	Molten Salt cooled Fast Reactor
MSR	Molten Salt Reactor
MTR	Materials Test Reactor
NEA	Nuclear Energy Agency
NEPA	National Environmental Policy Act
NES	Nuclear Energy System
NEST	Nuclear Education, Skills and Technologies Framework
NEXIP	Nuclear energy X innovative promotion
NFC	Nuclear Fuel Cycle
NHSI	Nuclear Harmonization and Standardization Initiative
NPP	Nuclear Power Plant
NRC	Nuclear Regulatory Commission
NRG	Nuclear Research and consultancy Group
NRIC	Nation Reactor Innovation Center in USA
NSSS	Nuclear Steam Supply System
NTA	Normal Test Assembly
O&M	Operation and Maintenance
OCC	Operational Chemical Cleaning
ODS	Oxide Dispersed Steels
OECD	Organization for Economic Co-operation and Development

OGDHR	Operation Grade Decay Heat Removal
P&T	Partitioning and Transmutation
PCM	Phase Change Material
PDEC	Pilot Demonstrator Energy Complex
PFBR	Prototype Fast Breeder Reactor
PGSFR	Prototype Generation IV SFR
PHRS	Passive Heat Removal System
PIE	Post-irradiation Examinations
PIV	Particle Image Velocimetry
PRA	Probabilistic Risk Assessment
PRK	Point Reactor Kinetics
PSA	Probabilistic Safety Assessment
PSGIS	Prototype Fast Breeder Reactor Steam Generator Inspection System
PUREX	Plutonium Uranium Reduction Extraction
PWR	Pressurised Water Reactor
R&D	Research and Development
RANS	Reynold-Averaged-Navier-Stokes
RCVS	Reactor Core Viewing System
RTA	Rabbit Test Assembly
RVACS	Reactor Vessel Auxiliary Cooling System
SA	Subassembly
SAIGA	Severe Accident In-pile experiments for Generation IV reactors and Astrid project
SAGNE	Standing Advisory Group for Nuclear Energy
SAMG	Severe Accident Management Strategy
SC	Steering Committee
SCE	System Design Criteria
SCRAM	Safety Control Rod Axe Man
SDDS	SHADOC-based Design Development System
SDG	Safety Design Guidelines
SDGs	Sustainable Development Goals
SFR	Sodium cooled Fast Reactor
SG	Steam Generator
SGTR	Steam Generator Tube Rupture

SmART	Small Amount of Reused fuel Test
SMR	Small Modular Reactor
SMFR	Small Modular Fast Reactor
SMSFR	Small Module Sodium cooled Fast Reactor
SNF	Spent Nuclear Fuel
SPL	Simplified Spherical Harmonics
SPO	Solid-Phase Oxidation
SPX	SuperPheniX
SS	Stainless Steel
SSC	Structure, System and Component
SSDHRS	Secondary Sodium based Decay Heat Removal System
SSP	Secondary Sodium Pump
SST	Shear Stress Transport
SSTAR	Small, Sealed, Transportable, Autonomous Reactor
STEM	Science, Technology, Engineering and Mathematics
TDICT	Time Dependent Inlet Coolant Temperature
TH	Thermal Hydraulics
THETA	Thermal Hydraulic Experimental Test Article
TRISO	TRistructural ISOtropic particle fuel
TRU	Trans-Uranium element
TWG	Technical Working Group
UCTD	Upgraded Chen and Todreas Detailed
ULOF	Unprotected Loss of Flow
UPB	University Politehnica of Bucharest
UPIT	University of Pitesti
YGE	Young Generation Event
VHTR	Very-High-Temperature Reactor
VTR	Versatile Test Reactor
VVER	Water-Water Energetic Reactor
WP	Work Package
XS	Cross Section

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