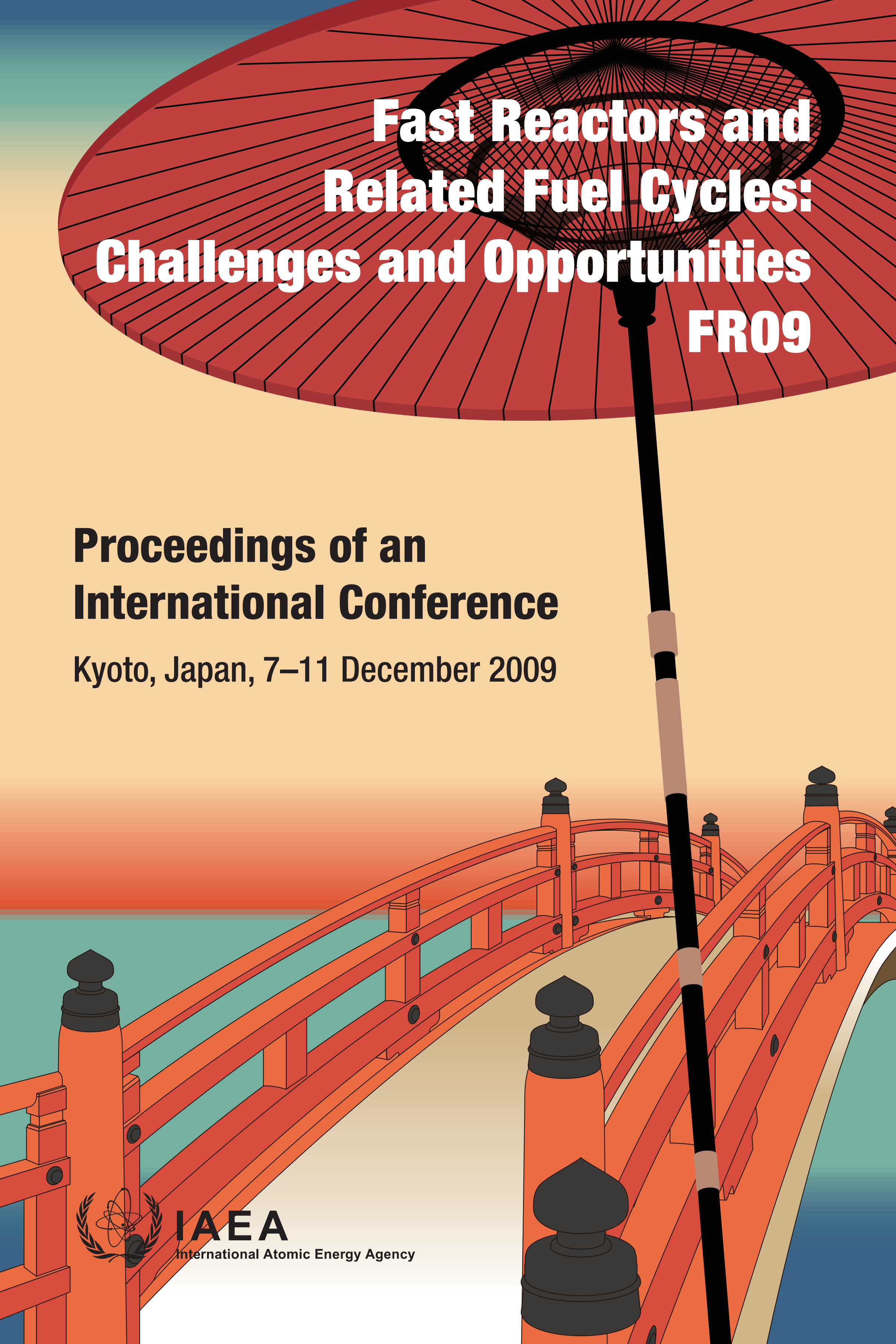

Fast Reactors and Related Fuel Cycles: Challenges and Opportunities FR09

Proceedings of an International Conference

Kyoto, Japan, 7–11 December 2009

IAEA
International Atomic Energy Agency

FAST REACTORS AND RELATED FUEL CYCLES: CHALLENGES AND OPPORTUNITIES (FR09)

The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world''.

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FOREWORD

Renewed interest in nuclear energy is driven by the need to develop carbon-free energy sources, by demographics and development in emerging economies as well as by security of supply concerns. It is expected that nuclear will provide vast amounts of energy in both emerging and developed economies. However, acceptance of nuclear energy with large scale contributions to the world's energy mix depends on satisfaction of key requirements to enhance sustainability in terms of economy, safety, adequacy of natural resources, waste reduction, non-proliferation and public acceptance.

Fast spectrum nuclear reactors with recycle significantly enhance the sustainability indices. The fast spectrum allows increasing the energy yield from natural uranium by a factor of sixty to seventy compared with thermal reactors, theoretically extending nuclear power programmes for thousands of years as well as significantly improving nuclear waste management. It is for this reason that fast reactors and associated fuel cycle research and technology development is, in many countries, back on the agenda of research and industrial organizations, as well as academia.

The way forward is tied to clear objectives, leading to the commissioning of experimental fast reactors (CEFR in China in 2010), the restart of the industrial prototype (Monju) in Japan in 2010, the commissioning, around 2012–2013, of power fast reactors in India and the Russian Federation (PFBR and BN-800, respectively), the planned construction, around 2020, of the French prototype fast reactor ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) and further advanced demonstration and commercial fast reactor construction projects in 2020–2050 in China, Europe, India, Japan, the Republic of Korea and the Russian Federation.

For more than 40 years, and in fulfilment of its statutory functions as outlined in Article III.A.1 – 3, facilitating research and technology development at the IAEA is being achieved through the mechanism of the Technical Working Groups, specifically, in the case of fast reactor and corresponding fuel cycle research and technology development, the Technical Working Group on Fast Reactors (created in 1967 as the International Working Group on Fast Reactors) and the Technical Working Group on Nuclear Fuel Cycle Options. The main aim of the Technical Working Groups is to provide a forum for exchange of non-commercial scientific and technical information and for international cooperation on generic research and technology development projects, and to enable scientists and engineers from research centres, industry and academia to share best practices globally.

In response to the expressed needs for an appropriate forum to achieve the twin objectives of exchanging experience and innovative ideas among experts, and of sharing knowledge and mentoring, the IAEA, after almost 20 years since the last large international conference dedicated to fast reactors and their fuel cycle, convened on 7–11 December 2009 in Kyoto, Japan, an International Conference on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09). The high expectations and interest in the conference were confirmed by the record attendance of 622 experts from twenty countries and three international organizations. This publication represents the proceedings of the conference. A CD-ROM of contributed papers accompanies these proceedings.

The IAEA would like to express its appreciation to the Japan Atomic Energy Agency, the host of the conference, as well as to the members of the International Advisory Committee and of the International Scientific Committee.

CONTENTS

NATIONAL AND INTERNATIONAL FAST REACTOR PROGRAMMES (Plenary Session 2)

ADVANCED CONCEPTS AND COOLANT TECHNOLOGIES (Plenary Session 3)

SAFETY AND MATERIALS (Plenary Session 4)

FUELS AND FUEL CYCLES (Plenary Session 5)

RETROSPECTIVES AND ADVANCED SIMULATION (Plenary Session 6)

CLOSING SESSION

SUMMARY

Introduction

Renewed interest in nuclear energy is driven by the need to develop carbon free energy sources. The drivers are demographics and development in emerging economies, as well as by security of supply concerns. It is expected that nuclear energy will contribute significantly towards meeting the energy needs of both emerging and developed economies.

However, societal acceptance of large scale contributions made by nuclear energy will depend on satisfaction of key drivers to enhance sustainability in terms of economy, safety, adequacy of natural resources, waste reduction, non-proliferation and public acceptance. It is a well established fact that fast spectrum reactors with recycle enhance the sustainability indices significantly.

The necessary condition for successful fast reactor deployment in the near and mid-terms is the understanding and assessment of innovative technological and design options, based on both past knowledge and experience, as well as on ongoing research and technology development efforts. In this respect, the need for in-depth international information exchange is underscored by the fact that the last large international fast reactor conference was held as far back as 1991. Since then, progress in R&D, as well as in design, has not been reported in a coordinated manner, which has made planning and implementation of expensive research and technology development programmes rather difficult. Consequently, there is a perceived need for an appropriate forum to achieve the twin objectives of exchanging experience and innovative ideas among experts, and of sharing knowledge and mentoring, whereby experienced scientists and technologists, as well as fast reactor programme managers, would share their perspectives with the future generation of young scientists and technologists, helping them to choose research problems of eminence and pursue their careers to meet the challenges of the development of fast reactors with recycle. After a hiatus of 18 years, this is also the appropriate time, as fast reactor programmes are currently on an accelerated growth path in many countries around the world.

It is in light of these reasons that the IAEA convened on 7–11 December 2009 the International Conference on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09) in Kyoto, Japan. The conference, hosted by the Japan Atomic Energy Agency (JAEA), aimed at promoting the exchange of information on national and multinational programmes and on new developments and experiences, with the goal of identifying and critically reviewing problems of importance and stimulating and facilitating cooperation, development and successful deployment of fast reactors in an expeditious manner.

The conference was structured to provide a comprehensive review of the potential of the fast reactor and its associated fuel cycle vis-à-vis the aforementioned key drivers. Thus, its scope included the key scientific and technological areas (e.g. fuels and materials development, safety, advanced simulation, component and system design, coolant technology, fuel cycles) in which innovation is pursued to ensure that the next generation's fast reactors and related fuel cycles achieve their potential. The programme comprised an Opening Session, six Plenary Sessions, eighteen Parallel Sessions, a Poster Session, a Closing Session, two Panels, and two Special Events, the Young Generation Event and the Special Tsuruga Session. The conference was attended by 622 experts from twenty countries and three international organizations. There were 150 oral presentations (including six opening remarks and keynote talks, three closing statements, nine panel and eleven Young Generation presentations) and 154 posters displayed.

Opening session

The Opening Session comprised three welcome addresses and four keynote lectures.

The welcome addresses were by T. Okazaki (JAEA President), Y. Kawabata[1] (Minister of Education, Culture, Sports, Science and Technology) and Y. Amano[2] (IAEA Director General). Apart from welcoming the participants and acknowledging the efforts of all those involved in the organization of the conference, a common theme of the welcoming addresses was the observation that nuclear energy is regaining worldwide recognition as an indispensable part of the world's long term sustainable energy supply. All three addresses emphasized the important role of fast reactor technology in meeting nuclear energy's sustainability goals with regard to both resources utilization and waste management. From their respective standpoints, all three speeches provided the views of research organizations, government and international organizations with regard to the main drivers for the development and challenges of fast reactor technology, i.e. global warming, resource utilization, energy security, waste management, non-proliferation, economics and public acceptance. By the same token, the speakers underlined the importance of continued research and technology development to fulfil the requirements and meet the challenges of the fast reactor

[1] Read on behalf of the Minister by K. Hakozaki (Deputy Director General of the Research and Development Bureau of the Ministry).

[2] Statement delivered by video.

and its fuel cycle, and assured the audience of the respective organization's support.

The keynote lectures provided high level overviews of the challenges and opportunities of nuclear energy, and more specifically of the fast reactor technologies, to address the world's energy needs in the current context of low carbon emission imperatives, economic competitiveness, demographics and development in emerging economies, resources utilization and security of supply, as well as the back end of the fuel cycle. The lectures expanded on topics already touched upon by the welcome speeches and provided the perspective of policy makers, researchers and technologists (papers by S. Kondo, Chairman of the Japan Atomic Energy Commission; P.B. Lyons, Principal Deputy Assistant Secretary, Office of Nuclear Energy, US Department of Energy; and J. Bouchard, Advisor to the Chairperson of the Commissariat à l'énergie atomique (CEA)), as well as of regulators (paper by M.-P. Comets, French Nuclear Safety Authority (Autorité de sûreté nucléaire) Commissioner).

The importance of defining strategic goals and the corresponding research and technology development programmes to address them was highlighted by all keynote speakers. In implementing these national roadmaps, the importance of a staged approach to short, medium and long term activities was emphasized. Thus, both the Japanese and US approaches underline, in the short term, safe and economic operation of existing nuclear power plants, as well as their lifetime extension, while the mid-term and long term objectives address development of innovative (next generation) reactor and fuel cycle technologies, as well as nuclear energy applications extended to transportation and industrial sectors.

More specifically and related to the fast reactor and its fuel cycle, the overview of current and planned research and technology development activities undertaken to tackle the challenges and make the best use of the opportunities indicates that there is no 'single best' solution or approach for the selection of the coolant, fuel, or plant concept. On the basis of past experience and newly gained insights, pros and cons must be carefully evaluated and alternative solutions and concepts put forward. All speakers agreed that the tasks at hand make international collaboration to "leverage capabilities and share facilities" a must, since "nuclear technology development and demonstration are too costly for a single nation to fund alone"[3]. Moreover, international cooperation is also necessary for the realization of experimental, demonstration and prototype fast reactor construction projects currently considered within the framework of national programmes.

[3] P.B. Lyons, this conference.

The views of both the regulator and the technologist were also addressed in the keynotes.

The latter emphasized one more time the importance of, firstly, reaching "international consensus on common (or compatible)" safety and security principles and objectives, and secondly, of establishing international standards for safety and security, permitting "the establishment of reference regulatory practices and regulations"[4].

With regard to the French sodium cooled fast reactor project ASTRID currently planned for realization around 2020 (see next section), the current position of the French Nuclear Safety Authority, as presented in M.-P. Comet's keynote paper, envisages an iterative process to define the safety objectives based on in-depth analyses of national and international experience feedback.

Fast reactor programmes

The national fast reactor programmes of China, France, India, Japan, the Republic of Korea, the Russian Federation and the United States of America were presented during two plenary sessions, along with the international programmes in the European Union (EU), the IAEA and the OECD Nuclear Energy Agency (OECD/NEA).

China

China is about to meet the first essential milestone in its fast reactor technology development through the commissioning of the 65 MW(th) China Experimental Fast Reactor (CEFR). The conceptual design of the 600–900 MW(e) China Demonstration Fast Reactor is ongoing. The next concept, currently under consideration, leading to the commercial utilization of fast reactor technology around 2030 is the 1000–1500 MW(e) China Demonstration Fast Breeder Reactor. By 2050, China foresees an increase in its nuclear capacity up to the 240–250 GW(e) level, to be provided mainly by fast breeder reactors.

France

Fast reactor technology development activities are determined by two French Parliamentary Acts, i.e. the 13 July 2005 Act specifying the energy policy guidelines and the 28 July 2006 Act outlining policies for the sustainable management of radioactive waste, and requesting R&D on innovative nuclear

[4] J. Bouchard, this conference.

reactors to ensure that, firstly, by 2012 an assessment of the industrial prospects of these reactor types can be made, and secondly, that a prototype reactor be commissioned by 31 December 2020 (with an industrial introduction of this technology in 2040–2050).

To meet the stipulations of these laws, the CEA and its industrial partners (EDF and AREVA) are implementing an ambitious research and technology development programme aiming at the design and deployment of the 300–600 MW(e) sodium cooled fast reactor prototype (ASTRID).

Within the framework of EURATOM projects, the CEA is also pursuing conceptual design studies for a 50–80 MW(th) gas cooled experimental prototype reactor (ALLEGRO).

India

First criticality of the indigenously designed (by the Indira Gandhi Centre for Atomic Research) and constructed (by BHAVINI) 500 MW(e) Prototype Fast Breeder Reactor (PFBR), located at Kalpakkam, was planned for 2011. The next step foresees the construction and commercial operation by 2023 of six additional mixed uranium–plutonium oxide fuelled PFBR type reactors (a twin unit at Kalpakkam and four 500 MW(e) reactors at a new site to be determined). The design of these six fast breeder reactors will follow an approach of phased improvements of the first Kalpakkam PFBR design. Beyond 2020, the Indian national strategy is centred on high breeding gain reactors of circa 1000 MW(e) capacity and on the collocation of multi-unit energy parks with fuel cycle facilities based on pyrochemical reprocessing technology.

Japan

On the basis of the 2006–2011 Science and Technology Basic Plan, in which the Council for Science and Technology Policy of the Japanese Cabinet Office identified fast breeder reactor cycle technology as one of the key technologies of national importance, the Ministry for Education, Culture, Sports, Science and Technology defined the Research and Development Policy on Fast Breeder Reactor Cycle Technology.

The Japanese fast reactor design and deployment activities are expected to lead to the introduction of a demonstration fast reactor around 2025 and to the commercial operation of fast breeder technology around 2050. These goals will be achieved on the basis of operational experience to be gained with the prototype fast reactor Monju (restart foreseen in the first quarter of 2010) and on the basis of the results of the Fast Reactor Cycle Technology Development Project (FaCT), which started in 2006 and which will develop the innovative technologies aimed at economic competitiveness, high reliability and safety of the next generation fast breeder reactors.

Republic of Korea

The fast reactor development activities of the Republic of Korea are performed within the framework of the Generation IV International Forum. Currently, R&D activities are focused on core design, heat transport systems and mechanical structure systems. Specifically, R&D work covers a passive decay heat removal circuit experiment, S–CO_2 Brayton cycle systems, a Na–CO_2 interaction test, as well as sodium technology. Design work on innovative sodium cooled fast reactor and fuel cycle concepts is being pursued. The Republic of Korea is planning to develop and deploy a demonstration fast reactor by 2025–2028.

Russian Federation

The Russian Federal Target Programme for Nuclear Power Technology of a New Generation for the Period 2010–2020 aims at enhancing the safety of nuclear energy and resolving the spent fuel issues. The Russian Federation established a mid-term plan to concentrate on fast reactor technology without constructing new light water reactors. The existing light water reactors will continue to operate and their spent fuel will be used to fuel the next generation of fast reactors.

The Russian fast reactor programme is based on extensive operational experience with experimental and industrial sized sodium cooled fast reactors. The Russian Federation has also developed and gained experience with the technology of heavy liquid metal cooled (lead and lead–bismuth eutectic alloy) fast reactors. The Russian Federation is currently constructing the sodium cooled, mixed uranium–plutonium oxide fuelled BN-800 with planned commissioning in 2012–2013.

The fast reactor development programme includes life extension of both the experimental reactor BOR-60 and the industrial reactor BN-600 (the latter ended in April 2010), and the design of the new experimental reactor MBIR (100 MW(th)/50 MW(e), sodium cooled, uranium–plutonium oxide (alternatively uranium–plutonium nitride) fuelled), planned as a replacement for the BOR-60. Within the framework of the programme, fast reactor technologies based on sodium, lead, and lead–bismuth eutectic alloy coolants (i.e. sodium cooled fast reactor, BREST-OD-300 and SVBR-100, respectively) will be developed simultaneously, along with the respective fuel cycles. The design of the advanced large size sodium cooled commercial fast reactor BN-K is also ongoing.

USA

The former programmatic approach in the USA was centred on incremental improvement of existing technologies to allow for short term (~20 years) deployment of fast reactors. This was driven by the need to utilize the Yucca Mountain facility better. The challenges related to this approach and the corresponding choices of technologies and integrated systems were determined by the Yucca Mountain characteristics and project timescale (in other words by coordination with the national geological disposal strategy/plans). A notable consequence of this 'industrial' approach was that very limited investment was made in research and technology development in the actual innovation in the tools needed to develop a better understanding of the fundamentals.

The current US programmatic approach is centred on the long term deployment of fuel cycle technologies, the initial analysis of a broad set of options and on the use of modern science tools and approaches designed to solve challenges and develop better performing technologies.

One major goal of the US programme is to develop an integrated waste management strategy. The focus of this work is on predictive capabilities for understanding repository performance. Another major research focus is in the area of used fuel separation technologies. Through the use of small scale experiments, theory development, as well as modelling and simulation to develop fundamental understanding, innovative long term options are being explored. The aim of this work is waste reduction. Enhanced materials protection and control is another key goal in the US fast reactor programme. In this area, the work focuses on the development of advanced techniques providing real time nuclear materials management with continuous inventory (including for large throughput industrial facilities).

The following technical areas are recognized as challenging areas requiring enhanced R&D efforts: (i) safe and cost effective storage and disposal of used fuel, high level waste, greater-than-class-C waste and low level waste; (ii) recycling technologies and economic recovery of transuranics (TRU) for recycle/transmutation and (iii) advanced technologies for materials protection, accounting and control.

A variety of fuel cycle options will be investigated through the fuel cycle R&D efforts. For the closed fuel cycle, the advanced recycle reactor will be developed and demonstrated with the help of R&D, focusing on capital cost reduction, assurance of safety and high system reliability. The fast reactor system R&D and related activities are ongoing and progress has been made in the areas of concept development, nuclear data, advanced materials qualification (especially regarding manufacturing and inspection, and applicability of the

ASME codes and standards), advanced energy conversion systems, safety, transmutation fuels, as well as modelling and simulation.

EU

The EU's Strategic Research Agenda is well balanced among the following three domains: current and future light water reactors, Generation IV fast reactors and other applications of nuclear energy.

The EU's Strategic Energy Technology Plan, established in March 2008, includes strengthened R&D efforts for nuclear fission with fast reactor technology development as one of its initiatives. Fast reactor development is encouraged to ensure long term resources availability and to increase waste management efficiency. The focus of the EU's fast reactor development is on safety, operational aspects and competitiveness. The effort is implemented through the European Sustainable Nuclear Energy Industrial Initiative (ESNII), with the objective of developing and demonstrating the sustainability of Generation IV fast reactors. The target for deploying the first Generation IV reactor is 2040, with at least a 30% share of EU electricity generated from nuclear power. Many European industries are participating in ESNII. Within the ESNII framework, studies are performed for three fast reactor concepts, i.e. sodium, lead and gas cooled, respectively. For the planned prototype reactor, to be commissioned around 2020, the sodium cooled fast reactor is currently considered to offer the reference, proven technology, while lead and gas cooled reactors are considered to provide alternative technology options.

The EU is also pursuing fast reactor development activities through EURATOM and Joint Research Centre projects. Through various EU Framework Projects, EURATOM is supporting fast reactor research and technology development activities for sodium, lead and gas cooled reactors and for the Molten Salt Fast Reactor. The technical support is mainly provided by the Joint Research Centre, with contributions for R&D activities covering various technical areas, e.g. fuel properties, characterization and fabrication, advanced fuel reprocessing experimental studies, structural materials, reactor design and safety, as well as proliferation resistance and physical protection. The EU considers international R&D collaboration to be of primary importance and thus EURATOM takes active part in a number of bilateral and multilateral collaboration initiatives aiming at the development of Generation IV fast reactor technologies.

IAEA

Acceptance of nuclear energy with large scale contributions to the world's energy mix depends on the satisfaction of key drivers to enhance sustainability in terms of economy, safety, adequacy of natural resources, waste reduction, non-proliferation and public acceptance. Fast spectrum reactors with recycle significantly enhance the sustainability indices. Hence, fast reactor and associated fuel cycle research and technology development is, in many countries, back on the agenda of research and industrial organizations, as well as academia.

The IAEA's activities in the field of advanced fast reactor and associated fuel cycle research and technology development are implemented within the framework of various Technical Working Groups (TWGs), specifically the TWGs on Fast Reactors (TWG-FR), on Nuclear Fuel Cycle Options (TWG-NFCO), and on Fuel Performance Technology (TWG-FPT). Recognizing that technological innovation is a key for maximizing the benefit from the use of nuclear energy for sustainable development, and centred on the solid support of, and leverage offered by, the TWG communities, the IAEA has been offering over the past 40+ years to all countries wishing to pursue fast reactor development the only worldwide forum for information exchange and collaborative R&D. The activities implemented within the framework of the TWGs ensured that 'newcomers' such as China and India are not left in isolation, but have a forum through which to ask technical and scientific questions, exchange information and hone their skills (both in human resources and in methodologies, i.e. data and codes).

Looking ahead on a mid-term basis, worldwide developments are diverging, with one group developing the fast reactor by focusing primarily on its recycling capabilities (geared towards alleviating the problems of the back end of the fuel cycle), and another group looking at it primarily as securing energy needs while enhancing resource utilization by a factor of ~70. Both of these aspects are strongly responding to sustainability requirements. Consequently, the IAEA's Member States' activities over the next 20–30 years will focus on different objectives and produce a variety of results and experiences. A wide range of scientific and technical areas will have to be covered (e.g. nuclear data, reactor physics, engineering design, methods validation and qualification, fuel and material characterization and development, and fuel cycle strategies). By the same token, Monju in Japan and the fleet of six PFBRs in India will produce a wealth of physics and performance, as well as operational experience, data. Moreover, new experimental and prototype fast reactors will be built in France (ASTRID), the Russian Federation (MBIR) and possibly in the Republic of Korea and the USA. Within this context, it is expected that the IAEA will be asked to further enhance its international collaborative R&D and information exchange activities and play an important role in support of the development and realization (and later also utilization) of experimental and research facilities through international collaboration.

OECD/NEA

The role of the OECD/NEA is to assist its member countries to develop the scientific and technological bases for safe, environmentally friendly and economical use of nuclear energy. To this end, the OECD/NEA coordinates international projects involving experts from member countries.

The OECD/NEA does not have any comprehensive programme on fast reactors, but does, rather, provide various activities in support of fast reactor development in different fields, e.g. nuclear data, structural materials, fuels, reactor, as well as partitioning and transmutation. Moreover, the OECD/NEA is performing strategic studies, e.g. addressing transition scenarios from thermal to fast reactors, nuclear fuel cycle investigations, as well as availability and efficient utilization of experimental facilities. As a current activity, the OECD/NEA provides technical reviews of the MYRRHA project, an accelerator driven lead–bismuth eutectic cooled subcritical concept proposed by Belgium. Last, but not least, the OECD/NEA is fulfilling the role of technical secretariat of the Generation IV International Forum.

Major challenges of innovative fast reactor concepts

Two overview presentations summarized the driving forces determining innovative fast reactor core concepts and the major challenges facing liquid metal coolant technology development.

The first paper identified the two main categories of driving forces, i.e. waste management for the current large light water reactor (LWR) countries having modest nuclear energy growth rates and confidence in longer term uranium availability, and resources for countries with ambitious nuclear power deployment plans having fissile material breeding as the main concern.

The paper then gave an overview of the R&D efforts aimed at finding innovative solutions for the major challenges in the fields of sodium cooled fast reactor fuel development (minor actinide-bearing oxide, metallic, carbide and nitride fuels), reprocessing, core design and safety.

As far as general core design, the choice of loop versus pool concept seems less fundamental than the fuel design and the associated reprocessing technologies. For India, reaching the target 1.45 breeding ratio and a 9-year doubling time implies implementing advanced metallic fuels after 2020. France and Japan are studying innovative aqueous reprocessing routes in view of their efforts to develop an advanced fuel cycle that includes minor actinide reprocessing.

In the safety design area, an important driving force is the prevention of recriticality, and to this end Japan is developing the FAIDUS device to enhance molten fuel discharge in severe accident situations.

The second paper reviewed the challenges linked with the utilization of liquid metals (both light (sodium) and heavy liquid metals (lead and lead–bismuth eutectic)). The main thrust of the paper was on mass (impurities) transfer and purification technologies. Taking SS316 steel as example, it was shown that the steel dissolubility is directly linked to the inclusion of oxides. Corrosion of this material is three to four times higher in heavy liquid metals than in light metal coolants.

The paper also discussed the accumulation of impurities and the consequent deterioration of the coolant's heat transfer characteristics. For oxygen, hydrogen and nitrogen, it is recommended that cold traps be employed, while carbonic traps are the best choice for caesium impurities.

Advanced and innovative reactor concepts

Contributions from France, Italy, Japan, the Republic of Korea and the Russian Federation presented the main features of sodium, heavy liquid metal and gas cooled advanced/innovative fast reactor concepts, as well as the driving forces behind them and objectives of the respective development programmes.

From a utility (EDF, France) point of view, the improvement of safety, availability, in-service inspection and repair (ISI&R), as well as construction cost reduction, are the main driving forces behind France's next generation sodium cooled fast reactor development programme. The objectives of its research and technology development activities are defined on the basis of a thorough review of the feedback from Phénix and Superphénix design, construction and operation.

France is implementing an important R&D programme to evaluate the industrial prospects of sodium cooled fast reactors for used fuel transmutation. In presenting the main design features of such a fast reactor concept, the need to satisfy the following requirements was emphasized: (i) robust safety demonstration for the core and the reactor systems (including prevention and mitigation of sodium risks and severe accidents, as well as robustness to external hazards), (ii) financial risk at the same level as for other plants and (iii) flexibility of nuclear materials management.

In Japan, the development of the next generation of sodium cooled fast reactors is performed within the framework of the FaCT project, whose objective is the commercialization of the fast reactor cycle system. The JAEA contribution summarized the main design targets, the status of the Japan Sodium Cooled Fast Reactor (JSFR) design studies and related research and technology development efforts aimed at achieving economic competitiveness, reliability and safety. Significant progress was reported on the following topics: validation of the integrated pump with IHX option (risks of IHX tube vibration), in-vessel thermohydraulics (measures against gas entrainment risks), fuel handling machine (pantograph system tests), double wall steam generator concept (industrial feasibility) and ISI&R (development of an under-sodium viewer).

A recent review of the JSFR options led to confirmation of the essential features of its design, specifically, the two loops option and the high chromium option for the piping material. However, three design options have been determined to require further significant R&D before being confirmed: the 'hot vessel' concept (insufficient margins found at this stage), the double tube steam generator concept (feasibility of 37 m long tubes) and the ODS option for fuel cladding (still significant irradiations and post-irradiation examination necessary before implementation in a reactor full core).

The intention of utilizing Monju[5] as a hub for research and development activities on the road towards commercialization of fast breeder reactors and the associated fuel cycle technology was reiterated. The first phase after completing the startup programme will be devoted to the demonstration of reliable fast reactor power plant operation, with particular attention paid to mastering all aspects of sodium technology. The next phase will be devoted to demonstrating technologies aimed at improving fast reactor economics and further enhancing its safety characteristics.

The Republic of Korea's advanced fast reactor development objectives are aligned with the Generation IV International Forum technology goals (i.e. sustainability, safety and reliability, economics, proliferation resistance and physical protection). The Korea Atomic Energy Research Institute (KAERI) contribution summarized its advanced sodium cooled fast reactor concept (KALIMER) studies. The results of several design studies on core, cooling system layout and structural integrity analyses were presented for both the KALIMER-600 and KALIMER-1200 concepts. Moreover, the status of various research and development activities in support of the KALIMER concept (e.g. development of a supercritical CO_2 cycle and of an under-sodium viewing system) was presented. Finally, a comparative study of KALIMER-600 and PRISM summarized similarities (e.g. use of metallic fuel and seismic isolation characteristics) and differences (e.g. monolithic versus modular concepts).

The Russian advanced BN-1200 sodium cooled pool type fast reactor concept is derived from BN-600/800 technology. The core design is strongly directed by Russian regulation following the Chernobyl accident, requesting a nearly zero global void reactivity effect. This requirement leads to a very 'flat' core (0.85 m height) with an upper sodium plenum. The power output (1200 MW(e)) is limited by railway transportation limits for large components

[5] The prototype fast breeder reactor Monju was restarted on 6 May 2010 after more than 14 years shutdown.

(e.g. rotating plugs). The BN-1200 design incorporates some advanced features aimed at improving its competitiveness (e.g. in-vessel fuel storage during two cycles (no external sodium storage), bellows on secondary circuits pipelines, casing type steam generator, use of B_4C in shielding materials) and safety (specific DHR circuits with dip coolers operating in natural convection, passive absorber rods design (inserted automatically at 800°C), adoption of integrated primary sodium purification).

In the Russian Federation, considerable effort is also put into the design of the lead–bismuth eutectic cooled, UO_2 fuelled, two loop, 100 MW(e) modular fast reactor concept (SVBR-100). To facilitate transportation of factory-made parts, the SVBR-100 has an integral (‘mono-bloc’) arrangement of the primary circuit. It is designed as a long life core (7–8 year core lifetime without refuelling). At 495°C core outlet temperature, the overall predicted net efficiency is approximately 36%. The SVBR-100 is intended for regional power production. Therefore, close location to cities must be accounted for, requiring extremely high standards in terms of safety, self-protection and passive behaviour (e.g. passive decay heat removal with a two day grace period without exceeding admissible temperatures). The SVBR-100 is facing two major issues: corrosion of structural materials in the lead–bismuth eutectic and ^{210}Po formation from bismuth being irradiated by neutrons. The relatively low core outlet temperature is in response to the former issue, as is the adoption of an on-line system for oxygen concentration monitoring. Due consideration is given to the second issue, in terms of radioprotection measures, as well as to appropriate procedures and management of repair and maintenance works.

Design feasibility studies for a lead cooled fast reactor are performed within the framework of EURATOM programmes. The objective is to demonstrate the economic competitiveness and safety of the European Lead System (ELSY), a 600 MW(e) MOX fuelled[6], pool type reactor. ELSY has a compact primary system and all the internal components are removable. The fuel assembly heads are accessible for handling at ambient temperature. ELSY has no secondary circuits, spiral wound steam generator, and decay heat removal with isolation condensers, double wall water cooled dip coolers and air cooling of the reactor vault. With a core outlet temperature of 480°C, the overall predicted cycle efficiency is approximately 43%. The main issue faced by the ELSY design is linked to corrosion in lead. R&D is pursued to identify fuel cladding surface treatments that would overcome this problem.

As a longer term alternative, a 2400 MW(th) gas cooled fast reactor concept is being studied in France. The gas cooled concept is helium cooled (7 MPa

[6] Nitride fuel is also considered as an advanced option.

pressure), fuelled with refractory uranium–plutonium carbide clad with SiC–SiC composite and features an indirect combined cycle with He–N_2 in the intermediate gas circuit. With a core outlet temperature of 850°C, this concept has an overall cycle efficiency of approximately 45%. The main challenges faced by this design are related to fuel feasibility and decay heat removal in accident conditions (depressurization). The safety concept of this design is based on the 'defence-in-depth' approach and 'as low as reasonably achievable' principles and the analyses performed use a combination of deterministic and probabilistic approaches.

Summing up, key acceptability issues for next generation sodium cooled fast reactors are linked to economics and enhanced safety characteristics. Investment protection is also a strong utility requirement, leading to R&D efforts in the area of ISI&R. Innovative fast reactor concepts (including alternative coolants, i.e. heavy liquid metals and gas) are being studied, with, again, the main objective of demonstrating robust safety features and, at the same time, economic competitiveness.

Advanced and innovative component and system design

The rationale for research and technology development in this area is, once again, the need to make next generation's fast reactors economically competitive and enhance their safety characteristics. The obvious link between improved component and system reliability on the one hand and economics on the other, was highlighted by various contributions, which focused on the most important components and systems, i.e. primary vessel, grid plate, control rod drive mechanisms, primary pipes, top shield, main circulation pumps, heat exchangers, steam generators and fuel handling systems. Another common thread was the heavy reliance made by the various teams on a 40-year experience with, and feedback from, the design, construction and operation of fast reactors. Moreover, research and technology development in this area benefits from advances and innovation achieved worldwide, highlighting the importance of international collaboration. Major highlights of the contributions are summarized below.

After completion of the 500 MW(e) PFBR currently under construction, the Indian fast reactor development programme foresees the construction of three twin units (each comprised of two reactors of 500 MW(e) capacity) with improved economics and enhanced safety characteristics. Preliminary analyses were completed, R&D areas identified and the respective implementation strategy clearly defined, aimed at achieving significant capital cost reduction through innovation and new concepts for the grid plate, primary pipes, top shield, fuel handling system and main vessel design. Currently, a net material saving of ~25% in the reactor assembly components seems possible, leading not only to capital cost reductions of the improved reactor assembly components, but also to construction time reductions.

In a similar approach, component and systems development for the Russian BN-1200 advanced fast reactor design is making extensive use of the experience accumulated with BN-350 and BN-600. Innovative solutions aimed at improving economics and safety characteristics of various components (e.g. main circulation pumps, intermediate heat exchangers, autonomous and air heat exchangers, steam generators, control rod drive mechanisms and cold filter traps) are put forward, and the respective research and technology development programmes needed for their substantiation outlined.

Work aimed at improving the accuracy of design methodologies, and thus reducing margins adopted to account for conservative approaches, was presented by the JAEA. Given the large impact of the reactor vessel on fast reactor plant construction costs, the realization of a compact and simple reactor vessel is one of the main objectives pursued within the framework of the FaCT project. The phenomena determining the thermal stresses and consequent mechanical loads on the reactor vessel and core support structures are complex. Accordingly, standard design approaches are relying on conservative assessments that make costly reactor vessel wall protection systems necessary. With the objective of improving accuracy and reducing margins, the JAEA is proposing novel thermal load modelling, an improved inelastic design analysis method, and a novel strength evaluation method at high temperatures.

One of the most innovative approaches pursued by the FaCT project to achieve economically competitive fast reactor designs is the development of an integrated intermediate heat exchanger/primary sodium pump concept. The status of this development work, in particular the results of tests carried out with a 1/4 scale model, was presented in a JAEA paper. The tests included vibration tests and gas entrainment tests. On the basis of the results of theses tests, design changes are being adopted, aimed at further enhancing the reliability of this component. Remaining unresolved issues requiring further research and technology development work include: improvement of analysis accuracy, monitoring of shaft vibration and design improvements to the ring plates to prevent gas entrainment.

Another important JAEA contribution addressing efforts aimed at the improvement of the analytical methods (and thus contributing to uncertainties reduction) summarized the status of the development of the flow induced vibration evaluation methodology in the primary cooling pipes of the JSFR. Theoretical efforts to develop this methodology are backed by experimental programmes and a systematic validation and qualification effort. The analysis of results of experiments investigating unsteady hydraulic characteristics in short elbow piping and performed using 1/3 scale and 1/10 scale single-elbow test sections for the hot-leg pipe indicated no effect of pipe scale. The 1/3 scale experiments have a less significant effect of swirl flow on pressure fluctuation on the pipe wall. The 1/8 scale experiments have revealed no clear flow separation in a larger curvature elbow case than that of the JSFR. For the cold-leg pipe experiments, the 1/15 scale experiment with double elbows has clarified that flow in the first elbow influenced flow separation behaviour in the second elbow. The applicability of the U-RANS, LES and DES approaches in the numerical simulation was confirmed by comparison with the 1/3 scale hot-leg pipe experiment. In a next step, the numerical results will be provided as input data for the structural vibration evaluation of the piping. The effect of inlet conditions will also be investigated for the hot-leg pipe experiments and 1/4 scale cold-leg pipe experiments will be carried out, as well as further R&D efforts involving the small scale experiments and simulation.

The original pool/loop comparison study performed during the JAEA fast reactor feasibility study concluded that for the JSFR, the material amounts were about the same for the pool and loop configurations. In a new JAEA pool/loop comparison study focusing on economics aspects, the review of the reactor vessel diameters of various pool concepts showed that the JSFR pool design had the smallest reactor vessel diameter. In this new study, it is therefore concluded that, as a loop, the JSFR design is economically competitive.

Innovative approaches in the design of the fast reactor fuel handling system, if successful, have a considerable potential to achieve construction and operating cost reductions. In France, as a result of the extensive experience and significant expertise gained in sodium cooled fast reactor design, CEA, AREVA and EDF are studying novel technical options and solutions to improve sodium cooled fast reactor fuel handling systems. Various concepts were evaluated and future R&D plans outlined.

An important element of the fast reactor fuel handling system is the used fuel washing and leaktightness testing step. Improvements achieved in the BN-800 design (as compared with the BN-600 design) allow increasing the efficiency of the system by a factor of two. Ongoing R&D in the Russian Federation is focussing on improving the characteristics of the radioactivity control sensors in the leaktightness test: development of such sensors that do not depend on the nitrogen gas parameters (e.g. pressure and moisture content) would considerably increase the reliability of the leaktightness test and ultimately result in cost reductions.

In Japan, advanced component design activity is pursued within the framework of the 4S (Super-Safe, Small and Simple) fast reactor development programme as well. The primary circulation pump of the 4S reactor is a large diameter high temperature sodium immersed electromagnetic pump, which (being a static device with only few moving parts) offers the potential advantage of long term maintenance free operation. Toshiba presented progress achieved in the realization of this advanced component, i.e. the design and fabrication of a prototype electromagnetic pump that has the same size as that of the actual 4S. The electromagnetic characteristics in air were measured and found to match the design specifications. A sodium test loop has been built and pump characteristics tests in sodium are ongoing.

Advanced fuel design and performance

Numerous activities related to the design and development of advanced high performance nuclear fuels are being conducted for future fast reactors in several countries. These developmental efforts include both technical and economical goals. Future fuels are expected to improve reliability and performance, maximize output and minimize fuel cost. Moreover, fissioning of plutonium and minor actinides (Np, Am, Cm), reducing radiotoxicity and improving proliferation resistance are the long term goals of the design of advanced fuels.

During this session, irradiation tests, fuel fabrication technology development and out-of-pile studies such as fuel property investigations were reported and discussed.

In Japan, the future fast reactor and its fuel cycle system, under development within the framework of the FaCT project, is expected to include oxide fuel with simplified pelletizing fabrication technology as a reference concept. Its driver fuel consists of large diameter annular fuel pellets, oxide dispersion strengthened ferritic steel cladding fuel pins with a ferritic–martensitic (F/M) steel subassembly wrapper tube and minor actinide-bearing oxide fuel. The target burnup of the driver fuel is 150 GW·d/t in discharge average, which corresponds to 250 GW·d/t peak burnup and 250 dpa peak neutron dose. Fuel development efforts, including out-of-pile studies such as material characteristics, experimental evaluation and fuel property measurements, irradiation tests and fuel fabrication technology developments, were planned and are in progress. Moreover, future fuel concepts will be widely characterized through Joyo irradiation tests and Monju demonstrations.

Different types of fuel such as mixed oxide PuO_2, UO_2, (pellet, vibro-packed), $UPuO_2$ (pellet, vibro-packed), UC, UN, UPuC, UPuN, oxide, nitride and carbide inert matrix fuels, and alloyed and non-alloyed metallic fuels have been investigated in the Russian Federation through the BN reactors operation. Moreover, experiments with UPuN, MgO and ZrN based fuels have recently been completed in the experimental sodium cooled fast reactor, BOR-60.

In order to meet the Generation IV requirements such as sustainability, proliferation resistance, waste management, safety and economics, important R&D efforts for the commercial BN-K reactor are currently under way. As a reference fuel, MOX fuel is considered as being more of a long term option than the nitride is assumed to be. Other types of dense fuel are also planned to be investigated.

In the USA, considerable efforts are invested in the development of advanced fuels for transmutation applications as part of the fuel cycle R&D programme. The main challenge for the development of transmutation fuels originates from the goals of achieving high burnup, operating at higher temperatures and incorporating minor actinides into the fuels. High burnups will allow uninterrupted reactor operations over longer periods of time and consequently reduction of spent fuel volumes and, eventually, fuel cycle reduction costs. High burnups are, however, associated with physical limitations which are primarily due to the swelling of the fuel and oxidation of the cladding's inner surface, as well as the dimensional stability of core materials such as cladding and subassembly duct due to high fast neutron dose. Higher temperature operation also challenges the performance of cladding materials and hence advanced cladding materials are needed for high temperature operation. The irradiation performance database for mixed nitride fuels ((U,Pu)N) is substantially smaller than that for metal carbide fuels, and these fuels can be considered to be at an early stage of development relative to oxide and metal fuels. Compared with metal carbide fuels, mixed nitride fuels exhibit less fuel swelling and lower fission gas release; on the other hand, the problem of production of biologically hazardous ^{14}C in nitride fuels fabricated using natural nitrogen poses a considerable concern for spent nitride fuel waste management. However, the interest in nitride fuel remains due to the combination of high thermal conductivity and high melting point.

Research and development of minor actinide-bearing fuels has made significant progress in Europe, with a number of scoping irradiation tests made on several candidate fuels for fast reactors and dedicated minor actinide transmutation systems (e.g. the accelerator driven system). Despite the convergence on MOX fuels, large R&D initiatives were dedicated to carbide and nitride fuels. Both homogeneous and heterogeneous concepts for minor actinide reactor recycling are considered. In the former, the minor actinides are added in small quantities to the MOX fuel, while in the latter, the minor actinides are loaded in significant quantities in UO_2. Irradiation programmes to test these concepts for pellet and sphere-pac fuel configurations are under way.

Fast reactor fuel cycle

A wide range of nuclear fuel cycle scenarios and projections have been analysed and presented, demonstrating the great flexibility of fast reactors, which can operate to breed fissile material, satisfying the goals of sustainability and resource utilization, and as burners in order to reduce inventories of TRU resulting from a previous deployment of nuclear reactors or to stabilize and reduce the minor actinide inventories. With regard to the latter points, many system problems of large scale power engineering relate to high and continuously growing volumes of spent nuclear fuel and a limited raw material base of existing nuclear power engineering with thermal neutron reactors.

In the Russian Federation, to solve these problems, the Federal Target Programme Nuclear Energy Platform was adopted in the summer of 2009. Within the framework of this programme, transition to a new technological platform of the Russian nuclear power industry is to be provided. The programme is based on the transition to a uranium and plutonium based closed fuel cycle with Generation IV fast neutron reactors. A MOX fuel fabrication facility for fast reactors is currently designed to start with plutonium recovered from Russian pressurized water reactor (WWER) spent fuels. A large scale reprocessing plant for LWR spent fuels is planned to start operation around 2025. Further, technologies for electrowining and vibro-packed fuel fabrication are being developed and irradiation tests for fabricated fuels are being carried out using the experimental reactor BOR-60 and the power reactor BN-600. Developed on this base, innovative nuclear power engineering is foreseen to fulfil Russian demands for energy resources for a historically observable time period, simultaneously solving the problem concerning reuse of earlier accumulated spent nuclear fuel.

France too has come to the realization that the deployment of LWRs with the current breeding ratios is not sustainable in the long term. Conventional uranium resources would be consumed before the end of the century and already engaged around the mid-century if the nuclear fleet were only built up with LWRs. Even taking into account unconventional uranium, the resources would be engaged before the end of the century. Fast reactor deployment would thus be an answer to the resources issues for a long term development of nuclear technology.

For countries favouring the continuity of nuclear energy development towards the deployment of new fast spectrum systems, one of the main priorities for such future fast spectrum systems is the minimization of waste which would necessitate the recycling of actinides and therefore implementation of their partitioning (specifically or grouped). Partitioning and transmutation technologies allow meeting the objectives of countries in Europe and elsewhere which have decided or envisaged different nuclear power and fuel cycle policies. In France, based on the waste management act of 28 June 2006, the R&D of partitioning and transmutation technology has been executed, aimed at the evaluation and a decision on fuel cycle technology being made in 2012. The MOX fuel fabrication facility known as ‘AFC’ (for the ASTRID core) is planned to be constructed before 2020. Further, the minor actinide-bearing fuel fabrication facility known as ‘ALFA’ is under consideration to demonstrate the minor actinide transmutation capability of ASTRID.

As for the current Japanese fuel cycle, plutonium recovered through reprocessing LWR spent fuels is currently used in LWRs, which will continue until the fast reactor cycle technology has been successfully developed in the future. The five party coordinate council consisting of Japanese Government authorities (Ministry of Education, Culture, Sports, Science and Technology and Ministry of Economy, Trade and Industry), utilities, vendors and the JAEA was established in July 2006 and is now discussing how to shift smoothly from the existing LWR cycle to fast reactor cycle.

As determined by the feasibility study on commercialized fast reactor cycle systems, executed from July 1999 to 2005, the main concept to be pursued will consist of a combination of a sodium cooled fast reactor accompanied by advanced reprocessing and simplified pelletizing fuel fabrication. Minor actinide-bearing MOX fuel is the main concept (presently the most promising technology) whilst electrorefining and injection casting of minor actinide-bearing metallic fuel is an alternative (as it permits improved core performance). Design studies and R&D of the main concept have been advanced through the FaCT project, which began in 2006, aided by data and experience collected from Joyo and Monju. It is foreseen that the conceptual design of both demonstration and commercial facilities, as well as the R&D plans for their practical use, can be presented in 2015, and a demonstration fast reactor and its fuel cycle facilities can start operating around 2025, with a commercial fast reactor introduced before 2050.

The USA announced the Global Nuclear Energy Partnership in 2006, in which R&D has been promoted, aimed at the commencement of commercial operation of an advanced recycling reactor and a consolidated fuel treatment centre that enables actinide recycling in the 2020s. Assuming that plutonium extracted by reprocessing may pose proliferation issues, a plan for commercial reprocessing has been frozen and fast reactor development has been prolonged for plutonium use since 1977. The development promotion of the nuclear fuel cycle technology and next generation nuclear power technology in the national energy policy was announced in 2001. The Advanced Fuel Cycle Initiative (AFCI) began in response in 2003. Although the development of these plants was frozen due to new nuclear policy shifting its direction towards scientific long term R&D, the AFCI is still ongoing as an advanced fuel cycle and waste management technology with proliferation resistance, aimed at the minimization of nuclear waste. Major issues reported in the AFCI are:

(a) Intermediate term issues associated with spent nuclear fuel, specifically reducing the volume of material requiring geological disposal by extracting uranium and reducing the proliferation risk through the destruction of significant quantities of plutonium contained in the spent nuclear fuel.

(b) Long term issues associated with spent nuclear fuel, specifically the development of fuel cycle technologies that could sharply reduce the long term radiotoxicity and long term heat load of high level waste sent to a geological repository.

In addition to descriptions of the national approaches to the nuclear fuel cycle involving fast neutron systems, several international perspectives were provided. These contributions analysed, at a global level, various transition scenarios and identified some of the areas and means by which international collaboration can be of interest in the area of fuel cycle development.

In a study from the Karlsruhe Institute of Technology in Germany, several scenario analyses in which 'breeder', 'isogenerator' and 'burner' fast reactors have been deployed with very different missions have been performed in order to investigate the wide range of applications associated with fast reactor deployment. In the case of breeders, the requirements of sustainability can be dealt with by making appropriate design choices (e.g. fuel type) in order to reduce the doubling time (e.g. below 10 years). In the case of burner fast reactors, they can be easily adapted to the assigned mission within a specific national or regional policy, by reducing and successively stabilizing minor actinide inventories or by drastically reducing legacy inventories of TRU, both in the extreme cases of an LWR only or fast reactor only power fleet deployment. This last point shows concretely that the choice of a fast reactor fleet, with TRU recycle, can be reverted and that the existing TRU inventories at a specific moment can be destroyed in a few decades by the same fast reactors, having converted them from breeders (or isogenerators) to burners.

The demonstration of possible synergies between nuclear energy deployments in different regions of the world were presented in a US paper, assuming hypothetical strategies for nuclear systems deployment that involve transitions to fast reactor systems. Dynamic simulation of multiregional nuclear energy deployment scenarios requires a complex analytical tool which has recently become available through an advanced version of the DANESS model. The scenarios consider transition to fast reactor systems with different conversion ratios according to possible forecast nuclear energy futures and strategies for waste minimization and resource sustainability. This sustainable nuclear energy deployment worldwide is investigated taking into account regional effects and synergies between the different regions. Simulation results show that converter fast reactors in a 'developed' region can be used to limit the inventories of TRU around the world. However, the reduction in TRU inventories can be limited if all converter fast reactors are built in the developed region using spent fuel from different parts of the world, especially if the developed region growth rate is relatively low (limits the possible deployment of fast reactors). This hypothetical study suggests that in order to reduce this effect, fast reactors will have to be deployed in other parts of the world as well. The deployment of small long-fuelling-interval fast reactors is also being considered. This is shown to help achieve the goals of limiting the TRU inventories and addressing sustainability issues.

The results were presented of the assessment of an innovative nuclear system based on a closed nuclear fuel cycle with fast reactors (CNFC-FR) that was performed jointly by Canada, China, France, India, Japan, the Republic of Korea, the Russian Federation and Ukraine within phase I of the IAEA International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). The joint study has served as an innovative, unique and cost and time effective multinational organizational structure to assess the role of upcoming and future nuclear energy systems. Multinational inputs have helped to confirm the important role of a CNFC-FR in a future global nuclear architecture as a key option for enhancing the sustainability features of nuclear power. The study has demonstrated an efficient complementary role for the IAEA Technical Working Groups (TWGs), i.e. the TWG on Fast Reactors and TWG on Nuclear Fuel Cycle Options, especially in focusing on the priority areas of pursuit, establishing collaborative projects of high relevance and presenting the unique and expensive facilities available in Member States which have mastered the fast reactor science and technology systems and, most importantly, for establishing cooperation in the frameworks of collaborative initiatives. The joint study has used a rational approach to define innovative R&D in priority areas of interest, identified the scope for improvements and demonstrated the readiness for enhanced collaboration, especially in the areas of safety and economics. A robust publication (IAEA-TECDOC) has been generated containing explicit and versatile knowledge on CNFC-FR systems. This publication makes essential contributions to a deeper understanding of CNFC-FR. The INPRO methodology was found to be a valuable instrument for a comprehensive analysis on how to enhance the sustainability features of nuclear power. Several recommendations and comments were made regarding its improvement.

Proliferation resistance and physical protection

This session focused on the proliferation resistance issues for the fast neutron reactors and the related fuel cycles. A large amount of plutonium will be necessarily handled in the future generation of fast reactors and related nuclear fuel cycle. To prevent the risk of nuclear proliferation, very robust measures for nuclear proliferation resistance will be needed and these have been presented by France, Japan and the USA. In particular, the analyses have been focused on the proliferation resistance of fast reactors by assessing the surronding issues and the efforts needed to identify reactor technologies and fuel cycles that are safer, more efficient in their generation of nuclear waste and more proliferation resistant.

Proliferation resistance measures comprise both intrinsic barriers or technical proliferation resistance of nuclear energy systems and extrinsic barriers or institutional barriers, for instance the application of improved safeguards and physical security protection systems that could bolster barriers to proliferation.

In this context, a new concept of differentiation in the intrinsic measures depending upon the level of safeguards could be applied from the viewpoint of plant design rationalization. Ongoing studies are focusing on the actual need for further analyses of these and other proliferation resistance measures, taking into account the ease with which they can be reversed or otherwise contravened by States, as well as the attendant safeguards costs.

Ultimately, the extrinsic barriers for proliferation resistance of these fuel cycles will depend on their ability to address safeguards challenges. For instance, at the reprocessing facilities, the key challenges include traditional safeguards issues such as measurement uncertainties in large bulk material handling facilities, the accuracy of plutonium measurements and process holdup inventories, as well as issues raised by the use of TRU fuels and other new developments.

Fast reactor safety

The contributions to this topical area focused on severe accidents and, in particular, on practical ways to eliminate core disruptive accidents (CDAs). The importance of prevention and mitigation of severe accidents was recognized, and hence of the need to depend on inherent safety features and/or design features making the severe accident progression benign. Ultimately, it is desirable to adopt a safety approach that would eliminate recriticality.

Fast reactor design has always recognized the importance of reducing the coolant void reactivity effect and adopting passive shutdown system features. Currently, there seems to be consensus that the deterministic approach will continue to be the guiding design principle to achieving a fast reactor and that a risk informed approach should be adopted for evaluating severe accidents.

On the practical way to eliminate CDAs, and considering the possibility of accelerated fast reactor deployment over the next decades, the short term objective is seen as lowering CDA probability figures as compared to the ones presently achieved. One contribution claimed that this can be achieved more easily in metal fuelled fast breeder reactor designs than in oxide fuelled ones. The discussion converged on the need to take into account recent R&D progress and perform (using the state of the art tools now available) comparative studies of the transient behaviour of small and large fast breeder reactor cores fuelled with different fuel types (oxide, carbide and metal).

Several contributions presented detailed analyses of various safety approaches and the specific R&D work being implemented by various groups.

The Japan Nuclear Energy Safety Organization contribution presented the technical guidelines for MOX fuelled fast breeder reactor safety evaluation, which cover both functional and structural integrity aspects identified under three major safety functions, i.e. shutdown, core cooling and containment. Safety evaluations were compared with the respective LWR criteria. The acceptance criteria were established based mainly on experiments, operating experiences, and in-pile and out-of-pile tests.

The JSFR safety design requirements for safety systems and components are based on CRBRP, PRISM, SPX, LWR and IAEA standards, as well as on the Generation IV International Forum safety targets. The requirements cover innovative core and fuel design features to achieve high burnups, longer operation time, mitigation of energy release under CDA, as well as safety features to restrict the sodium void effect (i.e. the incorporation of an inner subassembly duct for molten fuel discharge).

The French safety approach for future sodium cooled fast reactors is similar to that adopted for the JSFR, except for considering ex-vessel degraded core cooling as an option.

The Generation IV International Forum Risk and Safety Working Group contribution covered the Generation IV International Forum safety approach and objectives, defence-in-depth philosophy, risk informed design and assessment approaches, as well as modelling and simulation aspects. The integrated safety assessment methodology, for which work is ongoing, was described and its five objectives outlined.

Efforts to develop integrated analytical tools for level two probabilistic safety analysis of liquid metal cooled fast breeder reactors are ongoing in Japan. A JAEA contribution summarized both the severe accident scenarios and the analytical tools (e.g. ABAQUS, ASTERIA, AZORES) developed for their analysis. Phenomenological distribution diagrams were established, providing an analytical perspective that starts from the highest level and extends to the lowest level of severity. One important aspect that was highlighted is the generation of a database of the Monju emergency response system.

A contribution from Argonne National Laboratory presented results of uncertainty analyses for the unprotected loss-of-sink, loss-of-flow and transient over power events in sodium cooled fast reactors. The studies were performed for an 840 MW(th) sodium cooled advanced burner fast reactor. The paper provides a method that proves to be useful in a risk informed regulatory approach, since it allows estimating the probabilities for violating safety boundaries.

Structural materials

Contributions to the topic of structural materials summarized the present knowledge base and identified new challenges faced in the development of materials for the future generation of fast neutron systems. Materials requirements have evolved and objectives for the future generation systems must be able to withstand greater burnup (up to 20–30 at.%), longer operational lifetime (60–80 years), higher operating temperature (up to 700°C) and higher breeding ratios (up to 1.45) and be compatible with the transmutation of actinides. Future directions of research are focused on topics such as: the development, characterization and qualification of alloys; the improvement of codes to reflect improved modelling techniques and experimental validation; the development of new processes and modelling techniques in welding and inspection (including non-destructive) and the development of corrosion protection barriers.

Although individual countries may investigate unique solutions, the international trend converges towards 316 LN or 316 FR austenitic stainless steels for structural components (e.g. reactor vessel and internals), 9–12 Cr F/M steels for the wrapper and 9–12 Cr oxide dispersion strengthened steels as the clad. The combination of low thermal expansion and good thermal conductivity also renders F/M steels an attractive option for piping applications. There is also interest in 12–15 Cr oxide dispersion strengthened steels which, in addition to tolerating a high neutron fluence, allow corrosion during storage and reprocessing to be minimized. Characterization and modelling are showing better results relating to directions of research and assessment of performance. The materials for outside the core structures have convergence in materials with minor changes. Codification of materials for 60 years life and for different sizes, shapes and forming technologies is a current pursuit of manufacturing technologies, except for large thick forgings. Advanced non-destructive examination methods and in-service inspection are also being actively pursued.

Scientific based approaches to materials development, whose objective is to reduce the time required to develop steels with superior properties from tens of years to a few years, are currently being investigated. Such an approach involves the use of modelling, analytical systems and experience of industrial steel manufacturers in order to achieve the design goals. The approach has already been demonstrated to increase significantly the yield strength of 316 stainless steel simply by the use of a modified heat treatment procedure; the increased strength could allow size reduction and achieve longer lifetime and greater safety margins.

Some of the submitted contributions discussed the favourable irradiation performance (i.e. low or no swelling for irradiation up to about 160 dpa) of 12 Cr F/M steels. This confirms the promising future of F/M steels as core materials for the next generation of fast reactors. However, as pointed out, diverse operational practices would be required to be implemented, such as the intermediate storage of spent fuel assemblies, owing to their corrosivity in water. Other research seeks to develop F/M steels which have improved thermomechanical properties via modification of the impurity contents.

Other contributions addressed some of the key challenges associated with the actual development and deployment of advanced materials, with specific attention given to sodium cooled fast reactor systems. These key needs ranged from an initial assessment of alloy performance in elevated temperature mechanical testing or irradiation to analysis of the specific needs required for licensing. A common need in the testing of advanced materials for sodium cooled fast reactor applications is the assessment of irradiation performance. As evidenced by the contributions, particular attention has been given to swelling behaviour. Another common need for alloy development is code qualification and licensing of advanced alloys. On the basis of the presentations and discussions, there is a common recognition that advanced materials are important for improved reactor performance and economics. Resolution of code qualification needs and design methodology issues will be required for both existing and advanced materials. International collaboration is both beneficial and desired, and many countries already collaborate.

Research is advancing for materials used in heavy liquid metal systems as well. For example, the dissolution temperature limits for corrosion in heavy liquid metal systems of austenitic 316 steel and T91 F/M steel were identified and the problem of the enhanced oxidation rate of T91 F/M steel for temperatures above 450°C and its negative influence on heat transfer were discussed. This includes the interaction of the thermomechanical properties, such as the enhancement of the overall oxidation rate due to the hoop stress and the reduction of the creep strength due to contact with the liquid lead–bismuth eutectic. The very promising application of the surface alloying by pulsed electron beam (GESA) process creates a corrosion barrier which allows the temperature limits to be increased for steels employed in heavy liquid metal systems and improves the heat transfer capabilities and the creep strength of the steels. The entire GESA surface modification process to ensure a sufficient aluminium content has to be optimized and has to be further developed from the laboratory scale to the demonstration scale capable of producing actual components.

Coolant technology and instrumentation

Technology related contributions addressed the development of sodium purification systems, corrosion products mass transfer models, studies of alternative secondary loop fluids and experimental investigations of the wetting behaviour of sodium on various plated stainless steel surfaces.

On sodium purification, a contribution from China summarized the design, construction and commissioning of an indigenous sodium purification facility that is producing (starting from industrial grade sodium) the reactor grade sodium needed for the CEFR.

A French paper described the evolution of sodium purification system design from basic knowledge acquisition about the mechanisms and kinetics of crystallization, to innovative cold trap design efforts aimed at maximizing the purification rate, and to the development of regeneration processes (i.e. thermolysis for the decommissioning steps and an in situ process that avoids the dismantling of the cold trap).

Work on corrosion products mass transfer in sodium was reported in a comprehensive poster presentation from the Institute of Physics and Power Engineering (Russian Federation). The model takes into account the dissolution and crystallization of the impurities, their suspension and also their chemical interaction in sodium. The paper also reports results on model development work on steel oxidation in lead.

A French study evaluated various fluids (in addition to sodium) that are potential candidates as secondary loop fluids in a fast reactor, specifically three bismuth alloys, two nitrate salts and one molten hydroxide. The major criteria for the comparison with sodium were thermal properties, chemical reactivity with structures and with other fluids (air, water, sodium), chemistry control (including tritium management), safety and waste management, ISI&R, impact on components and circuits, as well as availability and cost. The study concluded that sodium, despite its reaction with water, remains the most interesting intermediate fluid when all the criteria are considered. Lead–bismuth eutectic is of some interest and should be further evaluated, since it raises a number of issues, such as corrosion of steel, which would require either lower operating temperatures or the development of new materials and, correspondingly, a lengthy R&D programme.

An experimental study on the wetting behaviour of sodium on plated stainless steel surfaces concluded that gold was the best plating element owing to its high solubility in sodium. It was also concluded that the wetting behaviour was dependent on the density of pinholes in the plating: the higher the density of pinholes, the worse the wetting properties, because of the oxidation of the steel surface through the pinholes.

The contributions in the instrumentation topical area covered sensor technology for on-line monitoring, liquid metal flow rates and velocity measurement techniques, and sodium leak detection.

On-line electrochemical sensors to monitor the sodium circuit for hydrogen, oxygen and carbon impurities, and diffusion based hydrogen meters coupled with semiconducting oxide based sensors to monitor hydrogen in the argon cover gas during startup and low power operation of the reactor are being developed at the Indira Gandhi Centre for Atomic Research in Kalpakkam, India. This paper summarized performance results of the sensors for on-line monitoring of hydrogen and carbon in sodium, and for hydrogen in argon covers gas circuits. Ongoing research activities to develop a sensor for monitoring oxygen in sodium were also addressed.

Measurements characterizing the liquid metal flow in a sodium cooled fast reactor (i.e. sodium flow rates, velocity distributions, turbulence intensity) are usually performed employing electromagnetic flowmeters. The contributions to this topical area covered various measuring techniques, summarizing application domains, advantages and drawbacks, remaining issues and R&D needed to resolve them. Each measurement technique has pros and cons, and the choice of the optimal method has to be made with regard to the actual experimental configuration and its parameters. Moreover, it is important to identify the kind of information that should be obtained from the measurement, as well as the desired spatial and temporal resolution and accuracy. Overall, it can be concluded that, for liquid metal flows at moderate temperatures, i.e. <300°C, a sufficient number of measurement techniques are available to characterize the flow, including phenomena such as solidification or two phase flows. Moreover, techniques are available and/or are being investigated to extend the higher values of the temperature range.

A German contribution presented three variants of fully contactless electromagnetic flowmeters for measuring integral flow rates in a channel, with one of the sensors not dependent on the electrical conductivity of the liquid metal and thus independent of the melting temperature. Ultrasound Doppler velocimetry provides the velocity profile along the ultrasonic beam and even has the capability to work through channel walls. The group developed an integrated ultrasonic sensor with an acoustic waveguide that can operate at temperatures up to 700°C. The paper also reported on the development of a method (employing contactless magnetic tomography of the mean flow in liquid metals) that yields the full three dimensional mean velocity distribution in a liquid metal volume. All development work reported is substantiated by experimental validation.

Two Japanese poster presentations reported sodium flow measurement technique developments for designs in which conventional electromagnetic flowmeter methods are not applicable due to material (e.g. ferromagnetic materials Mod. 9 Cr steel) and/or design (e.g. double walled pipes) choices, as well as to high temperatures and the radiation environment. The first poster reported on the development of a new type of electromagnetic flowmeter to measure flow at the annular flow pass of sodium components under high temperature and radiation regimes. The second poster summarized the status of the development of an ultrasonic flowmeter system for the safety protection system of the JSFR. Given the double wall piping foreseen, the ultrasonic transducers of this system are amenable to installation directly on the surface of the inner primary coolant pipes. This means that innovative design solutions were implemented to meet specific temperature, radiation level and remote handling requirements.

ISI&R

ISI&R for sodium cooled fast reactors is an area receiving heightened attention in all advanced reactor development programmes. This is due to the specificities of the sodium coolant and to the importance of advances in this area for the safety (verification of the state of the material and equipment over the whole reactor life is the first line of defence) and for the economics (reliable performance and investment protection) parameters of the next generation's sodium cooled fast reactors.

France is implementing a comprehensive R&D programme for the development of sodium cooled fast reactor ISI&R capabilities. This programme is based on the feedback from the operation of Rapsodie, Phénix and Superphénix and addresses primary circuit design improvements (aimed at limiting the number of structures and components to be surveyed, at providing space for remote control, at placing critical spots in areas that are accessible, and at reducing the number of welds), the development of measurement and inspection techniques (continuous monitoring during operation and non-destructive examination during maintenance periods), the development of remote control (robotics), and the development and validation of repair processes and techniques. The main milestones of this R&D programme are the experimental validation (in sodium) of the transducers (ultrasonic telemetry and sensors for non-destructive techniques), the identification of the key design requirements for the robotic equipment and the preliminary validation of repair processes and techniques (cleaning, machining, welding).

The JAEA is performing ISI&R research and technology development at its newly created FBR Plant Engineering Centre in Tsuruga. The R&D approach adopted is based on improvement of Monju technologies with the objective of achieving high performance and reliability of inspection, repair, replacement, leakage monitoring, and maintenance procedures for the commercial operation of fast breeder reactors. Currently, the JAEA considers time based maintenance and leakage detection to constitute the backbone of fast breeder reactor plant maintenance. However, it is recognized that achieving good performance and high reliability of commercial fast breeder reactor power plants, the introduction of condition based maintenance, as well as a sound combination of time based and condition based maintenance, will be needed.

For the JSFR, the ISI&R programme has been developed, keeping both the regulatory safety objectives and the investment protection objectives in mind. The repair programme is following the approach of the categorization of anticipated failures based on frequency and required repair level. The approach followed for the JSFR requires the consideration of ISI&R aspects already at the conceptual design stage. As a consequence, the design of major components has been improved and requirements for design changes and/or the development of additional devices were identified on the basis of the developed ISI&R programme.

Poster contributions reported on various innovative approaches to improve aspects of ISI&R. A Japanese group reported results obtained for a high sensitivity technique to detect defects in a helical-coil-type double wall tube steam generator (with a wire mesh layer between the two tubes). On the basis of experimental results, the group reported that it achieved an improvement of more than two orders of magnitude in the sensitivity of the remote field eddy current testing technique by means of increasing the indirect magnetic field intensity. This allowed the detection of a very small defect (1 mm in diameter extending over 20% of the outer tube thickness depth over the wire mesh layer). Another poster contribution reported the successful repair of 316L austenitic stainless steel plates with a slit type artificial crack, in both argon gas and liquid sodium environments, using the friction stir welding technique. On the basis of these developments, an in-vessel repairing machine concept has been established: the machine is inserted through the in-service inspection hole into the inside of the reactor vessel, where it is brought to the place of the repair with the help of a jointed robotic arm. A third poster contribution described the development of a new inspection robot for the ISI&R of Monju. The robot is using a tire-type ultrasonic sensor for volumetric testing at high temperature (55 and 80°C in air and on the pipe surface, respectively) and radiation exposure condition (10 and 15 mGy/h in air and on the surface of the pipe, respectively). The accuracy required in controlling the robot was very high, since the difference between the start and the end points after performing a complete run in circumferential and axial direction on the pipe surface was less than ±5 mm. In an automatic inspection test, an electric discharge machining slit having a depth of 10% of the tube wall thickness was detected. The signal to noise ratio obtained in this test was 4.0 (12.0 dB), largely exceeding the initial development goals (detection of a 50% depth slit with a signal to noise ratio equal to or larger than 2).

A JAEA contribution summarized the status of the activities in view of the planned restart of the Joyo experimental fast reactor. In-vessel visual inspection using the radiation resistant fibrescope and camera has been successfully conducted. The gamma ray dose rate measurement in the reactor vessel has provided useful information for the design of the shielding cask of the replacement above-core structure.

Twenty years' experience and fast reactor analysis

Efforts in the field of fast reactor analysis (basic data, experiments and simulation) were presented against the backdrop of past fast reactor design experience and the current sustainability drivers (costs, resources, waste management, safety) for, and challenges faced by, the development and deployment of advanced fast reactor and fuel cycle technology. Overview contributions from France, India, Japan and the Russian Federation, while taking pride in past achievements, highlighted the areas in need of further improvement, as far as availability, enhanced safety and cost and investment protection are concerned.

In the Russian Federation, establishing the technologies needed to transition to the closed fuel cycle is seen to be the major challenge.

France is relying on an impressive knowledge base gained from its past fast reactor programme to define the requirements – and perform the research and technology development activities required to meet them – of the next generation of sodium cooled fast reactors in the areas of core and fuel, structural materials, component design and sodium technology, as well as operation, maintenance and ISI&R.

Over the last 20+ years, India has made significant progress in the design and development of sodium cooled fast breeder reactors. These achievements are instrumental for the design of the next generation of fast reactors, for which enhanced safety and improved economics are the main objectives. Means to achieve the former are sought in elaborate ISI&R provisions, increased reliability of shutdown systems and decay heat removal systems, in-vessel purification system, and innovative post-accident heat removal provisions. Means to achieve the latter are sought in increased reactor lifetime (60 years), reduced construction time (5 years), increased plant load factors (85%), reduced special steel specific weight requirements and enhanced burnups.

Japan is also looking back on 20+ years of fast reactor development. In spite of setbacks (Monju and Joyo shutdown), the JAEA is committed to implementing its FaCT project and is confident of meeting the challenging safety, reliability and economic targets set for the development of the JSFR.

In shifting the attention from past design and analysis experience to simulation tools, an Argonne National Laboratory contribution reviewed research aimed at advanced simulation techniques for fast reactors. Looking back, the paper argued that existing fast reactor modelling tools were developed by engineers and physicists having in-depth knowledge derived from theory and underpinned by a vast repository of experimental data. Their general approach was to develop models that were tailored to varying degrees to the details of the reactor design, using free model parameters that were subsequently calibrated to match existing experimental data. The resulting codes were thus extremely useful for their specific purpose but, with the exception of neutronics, rather limited in their predictive capability. Looking ahead, the paper discussed current advanced simulation (science based) approaches, which aim at numerically solving the three dimensional physical equations and which allow observing and studying the emerging holistic phenomena. The most important potential benefit from this research lies in reducing uncertainties for existing fast reactor designs and enabling the exploration of more innovative designs with reduced reliance on physical experiments.

Highlights from the papers contributed to this topical area in three parallel oral sessions and a poster session are summarized below.

A Japanese group reported on an experimental study aimed at evaluating the possibility of positive reactivity feedbacks in the progression of hypothetical core disruptive accidents in fast reactors fuelled with metallic fuel. The design goal is to ensure that in the molten core, materials are passively discharged from the core region and the mechanism of thermal and hydrodynamic fragmentations due to the molten material–sodium coolant interaction was studied.

A European contribution reported on the results of a series of thermal hydraulics experiments performed at the Karlsruhe Liquid Metal Laboratory within the framework of the integrated project, Eurotrans. The experiments were performed in electrically heated and unheated hexagonal rod bundles in water and lead–bismuth eutectic. Their objective was to quantify and separate the phenomena of turbulent heat transfer and flow distribution in hexagonal rod bundles with the final goal of determining the momentum and the energy transfer in a heavy liquid metal cooled fuel assembly.

The objective of natural circulation tests performed at BN-350 was twofold, i.e. to demonstrate the reactor's stable decay heat removal capabilities and to obtain experimental data for code validation purposes. The analysis of the experimental results revealed a complex spatial behaviour of the natural convective flow. In addition to natural circulation phenomena due to non-uniform temperature distributions along the whole length of a loop, local natural circulation phenomena were also observed in certain sections of the loop.

A Japanese contribution reported on experimental work performed in a new sodium test loop facility aimed at demonstrating component performance for the Japanese 10 MW(e) 4S reactor concept. Plans were outlined for performance tests of a large electromagnetic pump, a failed heat exchanger tube detection system, as well as for reflector structural integrity. The goal of these tests is to demonstrate the high reliability of the 4S reactor concept.

Contributions from France and Japan reported on nuclear data related activities. Particular attention was devoted to the nuclear data needs of the analyst community and to the impact on this community's work. Papers and discussions can be summarized as follows: (i) nuclear data uncertainties are a major contributor to the uncertainty of reactor physics calculations, although, they are not properly taken into account in current fast reactor core neutronics analyses (e.g. the sodium void reactivity effect is calculated with an uncertainty amounting to ~2β_{eff}); (ii) another important source of uncertainties relates to the method used, while the diffusion theory approximation provides satisfactory accuracy in most cases, some problems (e.g. void reactivity effect calculations) require transport theory approaches; (iii) the new JEFF-3.1.1 nuclear data library has been validated against a number of critical and reactor experiments and is now released; (iv) a large OECD/NEA coordinated effort has produced an important outcome, i.e. the release of a new covariance library in 33 energy groups.

In the field of multigroup cross-section generation code development for fast reactor analysis, the status of the MC^2-3 code was presented. This work is being performed under the Nuclear Energy Advanced Modelling and Simulation programme of the USDOE. The MC^2-3 code integrates the MC^2-2 and SDX codes and implements various enhanced methods for resonance self-shielding and spectrum calculations. Development of efficient algorithms for in-line multigroup cross-section generation is in progress. Within the framework of verification activities, MC^2-3 was tested for several critical experiments. The MC^2-3 code is also being incorporated into the UNIC transport code to generate multigroup cross-section libraries consistent with the material and temperature distributions used in transport calculations.

The majority of contributions submitted to this topical area addressed method (code) development activities and applications. This is proof of the significant simulation development and validation work that is ongoing.

In terms of code development, the overall opinion is that the existing tools provide adequate means to predict the performance of fast reactor cores. It is noted that most core level simulation packages still rely on point kinetics neutronics and some form of a channel model in thermohydraulics. The structural modelling is also simplified, treating the complicated ducted bundle of wire wrapped pins as a simple beam. Given the availability of faster computing power, in comparison to several decades ago when most of these code packages were developed, it would not be surprising to see such future development work incorporate simulation capabilities in which many of these simplifications are removed. The overall picture offered by the contributions submitted to this topical area shows that some developers are focusing on enhancing the capabilities of existing tools, while others are rebuilding parts of their analysis packages or creating new ones. In all cases, it is noted that neutronics is considered to be generally satisfactory without the need for substantial changes. The major focus of the ongoing research is on the improvement of thermohydraulics modelling, such as the study of flow characteristics within the ducted hexagonal assemblies of wire wrapped pins in a sodium cooled fast reactor core under standard operating conditions and during blockage, as well as the development of whole system simulation capabilities for the design and safety analysis of sodium cooled fast reactors. Examples of specific issues addressed are:

— *Turbulent heat transfer*: An experimental and numerical study of turbulent heavy liquid metal (lead–bismuth eutectic) heat transfer along a uniformly heated rod within an annular cavity. The investigators studied the detailed momentum and energy distribution (average and time dependent) along this simulated fuel pin.
— *Thermal stratification effects*: A numerical study of the sodium thermal stratification effects in a sodium cooled fast reactor outlet plenum during loss of flow transients. The objective of this study was to improve the transient behaviour prediction of sodium cooled fast reactors under natural circulation conditions by taking into account multidimensional effects in the outlet plenum. The whole system simulation code SAS4A/SASSYS-1 was coupled to the 2-D/3-D outlet plenum model using the CFD code STAR-CD (one-way coupling as a preliminary study). This coupling method was applied to a transient simulation for the protected loss of flow. Results revealed the difference between the perfect mixing model and the multidimensional model for the outlet plenum, as well as the effect of the free surface model on the development of thermal stratification.
— *Turbulent diffusion*: An investigation of numerical thermohydraulics strategies for the simulation of wire wrapped sodium cooled fast reactor fuel bundles. Results of a comparative assessment of numerical simulation strategies using direct numerical simulation, large eddy simulation and Reynolds averaged Navier–Stokes analysis in subchannel analysis codes were presented. Comparisons of these results suggested that both approaches predict the hydrodynamic behaviour within the assembly with similar accuracy.
— *KALIMER-600 thermohydraulics analysis*: The multidimensional system code MARS-LMR (developed from RELAP5 and COBRA-TF) was used for a preliminary thermohydraulic analysis of the Korean fast reactor concept (KALIMER-600). Results for temperature and pressure distributions obtained for a 1-D model were compared with results obtained for 3-D models (with and without upper core internal structures in the hot pool).

Many contributions reported on R&D work performed to validate the existing methodology and codes against experimental data. In many cases, the existing suites of reactor analysis tools seem to perform very well, although further important developments are still possible and a number of national and international programmes addressing the development, validation, verification and qualification of advanced nuclear codes are being carried out.

Knowledge management and human resources

The mission of knowledge management is twofold: firstly, implementing programmes aimed at knowledge preservation (i.e. stop data and information from being destroyed, retrieve the data, assess their importance, determine what data and information should be retained, how information from different sources could be linked, how the quality of information should be assessed and establishment of software and hardware standards for preservation of the data), and secondly, passing information from one generation to the other. Obviously, the latter objective ties into human resources aspects.

There are several international initiatives aimed at knowledge preservation particularly in the field of fast reactors. An OECD/NEA Expert Group has published a report and created a web based database on Research and Test Facilities Required in Nuclear Science, including fast reactors in its scope. The OECD/NEA is also sponsoring databases that help preserve the wealth of information in various nuclear science and technology fields (e.g. the International Criticality Safety Benchmark Evaluation Project and the International Reactor Physics Benchmark Experiments). The IAEA has launched an international fast reactor knowledge preservation initiative which is implementing activities supporting digital document archival, exchange, search and analysis, and retrieval, as well as facilitating access to the information through a web based Fast Reactor Knowledge Portal. Numerous initiatives are aimed at knowledge transmission to the young generation (as also witnessed by the special session at this conference), e.g. joint IAEA/ICTP schools and workshops, the Frédéric Joliot/Otto Hahn summer schools on nuclear reactor physics, fuels and systems, and the World Nuclear University summer schools.

Reports from France, India and Japan also covered some national initiatives.

In France, the CEA established the Sodium and Liquid Metal School in 1975 and the Fast Reactor Operation and Safety School in 2005. The former is held at the CEA Cadarache Research Centre, the latter at the Phénix plant in Marcoule. More recently, courses implemented within the framework of the French National Institute for Nuclear Science and Technology, in collaboration with the Sodium and Liquid Metal School, specifically addressed Generation IV sodium cooled fast reactor topics. The syllabuses of the courses provided by the

CEA through the Sodium and Liquid Metal School and the Fast Reactor Operation and Safety School are comprehensive and cover sodium cooled fast reactor design, technology, safety, operation and decommissioning. The Sodium and Liquid Metal School, originally designed to meet national needs, is now open to foreign students.

The Indian contribution stressed the important role played by knowledge management in R&D organizations. This is particularly true in the nuclear energy area, which is characterized by long timescales, technological excellence and complex technology relying heavily on innovative creation, storage and dissemination of knowledge. Accordingly, the Indira Gandhi Centre for Atomic Research is implementing a management policy for creation, storage and dissemination of knowledge. With appropriate motivation schemes, complete involvement of employees and support from management, these efforts are successful. Ultimately, though, it must be recognized that the success of any knowledge management project depends on the “passion and profound belief that knowledge management is not only worthwhile but it is in fact a way of living”.

Contributions from Japan identified two national initiatives aimed at fast reactor education and training. The first one, a student training programme using the Joyo experimental reactor and related facilities, is in line with the JAEA’s mission of developing human resources for the nuclear industry and offers comprehensive syllabuses for students in the nuclear engineering and science departments of Japan’s universities. The second one introduced the International Nuclear Information and Training Centre. It implements various training programmes using the Fast Reactor Training Facility and the fast breeder reactor prototype Monju. The Fast Reactor Training Facility started in October 2000 and offers fast breeder engineering training courses. In addition, from 2006 on, the JAEA has hosted the Tsuruga Summer Institute on Nuclear Energy.

Maintaining and, when necessary, building new experimental facilities are essential, not only for advancing science and technology, but also for training and education. Contributions from Japan, the Republic of Korea and the Russian Federation reported on current initiatives in this area, i.e. Joyo core modification aimed at increasing the driver fuel burnup and thus enhancing the reactor’s irradiation capabilities, the design and construction by KAERI of a large scale sodium thermohydraulic test facility with the main objective of demonstrating the passive decay heat removal performance of advanced sodium cooled fast reactor concepts, and the programmes pursued in the Russian Federation to develop the experimental and material science base for the next generation of fast reactors, including the design of a new multifunctional experimental fast reactor to replace the BOR-60.

Young Generation event

Research and development of innovative technologies, conceptual design of commercial and demonstration fast reactors and fuel cycle facilities, and construction and operation of the facilities are essential for realizing the commercialization of the fast reactor and its related fuel cycle. It is expected, if not required, that the 'young generation' work diligently, passionately and cooperatively in these fields in order to successfully realize this goal. To that end, the Young Generation Event provided an opportunity for ten young generation representatives from around the world to respond to the following three questions in the form of a short presentation and then participate in a collective discussion with two senior professionals, directed by the panel moderator:

(1) What international cooperation is desirable with regard to fast reactor technology development and deployment to meet global nuclear energy sustainability requirements?
(2) What role do you expect for the IAEA in international cooperation with regard to fast reactor technology development and deployment?
(3) What international cooperation do you expect for the world's young generation?

The consensus of the discussions revealed undoubtedly that international cooperation and information exchange will play an important role to boost and advance fast reactor technology development and deployment in the world. Issues of present concern which would benefit include safety and economics. It was confirmed that communication, not only among the young generation but also between the young generation and senior professionals, is indispensable for the successful and sustainable development of nuclear energy through the development of human resources and the continuous transfer of technology and knowledge.

The young generation expect of themselves the establishment of an international network which will promote not only improved communication and friendships, but also provide encouragement and innovation. Effective cooperation among the young generation via organizations such as the International Youth Nuclear Congress and the young generation networks in each country should be established in the near future in order to share knowledge and experience.

Tsuruga session

The special session held in Tsuruga was organized to provide a forum for information exchange between experts and members of the general public, the result of which is anticipated to be improvement in the public acceptance of fast reactors. A total of 611 people participated in the session. Demographically, the majority (502 persons) were of Japanese origin, comprised principally of Tsuruga locals (478 persons); professionally, the majority (478 persons) were not registered FR09 conference participants, and comprised approximately 50% secondary and university level students (231 persons).

Presentations made to the public provided updated information on the significance, importance and major challenges, including issues of safety, economic competitiveness and non-proliferation of fast reactors on a worldwide basis. Questions posed by the public focused principally on an explanation of the differences between the fast reactor and the reference light water reactor and on safety aspects of the fast reactor system. Additional concerns, although not accommodated in the discussions, were presented in a questionnaire which was distributed to all participants of the Tsuruga session. Lessons learned from the questionnaires and discussions should contribute to understanding how to proceed in the area of public acceptance as regards fast reactor technology.

Panel on economics and performance of fast neutron systems: Overall reliability of plant and systems and impact of technological improvements

There have been great improvements and achievements in the design and engineering of fast reactor technology, both in overall reliability and in the safety of the systems. Utilities, vendors, R&D agencies and governments have been involved in technology development in their respective capacities and areas of expertise. In particular, engineering oriented work, rather than basic R&D, has led to great progress in improving economics as well as in enhancing safety. Studies in licensing issues and regulatory aspects are ongoing as well. Optimization of plant size and layout, reduction of the amount of plant materials and the building volumes, improvement of load factor, increase of burnup and replication of a series of reactors are good examples of accomplishments that improve economics.

The development of cost effective fast reactors, along with related fuel cycle technologies which have acceptable proliferation resistance, was identified as the main objective to be realized in order to complete the sustainable development of nuclear energy. To this end, and in light of the present situation, economics and proliferation resistance are two major challenges for fast neutron systems. The urgent challenge faced by plant designers, necessitated by the steep increase in capital cost, is to reduce the unit construction cost of a new nuclear power plant. This is especially necessary for those nuclear power plants using fast reactors; without demonstrating its economic incentives, fast reactor technology could potentially fail to become a viable option for sustainable energy development. Another challenge is to develop intrinsic design features of proliferation resistance which are in compliance with worldwide acceptance criteria. Doing so may cause an additional increase in the capital cost, which would negatively impact the economics of the fast reactor. Further effort to achieve the two objectives simultaneously is therefore required.

Above all, the importance of performing these R&D works through various close international or multilateral collaborations was acknowledged. The existence of such cooperation in all areas such as research, design and engineering, manufacturing and licensing would assist in dividing the otherwise high costs of developing the fast reactor systems. It was also suggested that fast reactor systems may benefit from international design evaluation and approval mechanisms such as the Multinational Design Evaluation Programme of the OECD/NEA or the Cooperation in Reactor Design Evaluation and Licensing working group of the World Nuclear Association.

Panel on international activities: Collaborative programmes, harmonization of prototypes, sharing of facilities and standardization

International collaboration and cooperation are important aspects for the worldwide development and deployment of fast reactors, as they provide a platform for the development of a long term nuclear strategy, the harmonization of R&D works, the sharing of experience and facilities, the enhancement of safety and standardization. A number of international collaborative programmes and multilateral cooperative projects currently exist, as well as various subprogrammes of other international consortia and bilateral or multilateral collaborations. New opportunities for international collaboration and sharing of facilities are constantly emerging.

Major challenges faced in this area include:

— Establishment of national nuclear strategy and political policy;
— Standardization in safety and design;
— Non-proliferation issues and verification system;
— Reduction of costs;
— Human capital, which includes education and training programmes, knowledge transfer to the young generation and funding.

Other challenges exist on a more programmatic level. Examples include the ability to implement international collaborative activities in the context of differing national programmes and the importance of performing complementary, rather than redundant, R&D in order to avoid duplication.

The panel concluded with the following recommended actions, to be implemented as required:

— Enhancement of international collaboration and cooperation through the further extension of regional and global collaborative programmes;
— Development of the regime in order to ensure fuel supply and fuel cycle services;
— Maintenance of an open dialogue in order to explore future opportunities for international collaboration.

Closing session

The importance of the fast neutron system originates from the initial Fermi intuition: whenever the option of sustainability and an optimized waste management is sought, a fast neutron spectrum is required. Opportunities for the fast neutron system are promising, yet challenges to realizing the full potential of such systems do exist. Reported results of the conference were encouraging, remaining open issues were identified and planned R&D programmes to resolve them were outlined.

Future challenges can be divided on the basis of timescale: short/medium term challenges and long term challenges. A major short/medium term challenge at this time is to allow for innovation despite the clear convergence on design options such as sodium coolant and oxide or metal fuels. Indeed, the sensible approach is to explore and develop a viable alternative option. Other medium term challenges include: availability and reliability, reversibility, convergence of safety approaches, fuel and clad performance, plant simplification and cost improvements, achievement of high conversion ratios while respecting non-proliferation concerns and reduction of uncertainties in all fields through the use of advanced simulation and validation experiments. It is reassuring to note that preliminary answers are available for many of these challenges.

Long term challenges at this time predominantly centre on fuel and materials. With the objective of simplifying the nuclear fuel cycle, it may be necessary to revisit the standard choice of solid fuel, inasmuch as it is related to reprocessing. Long lived cores are another possibility, but present materials challenges. It remains to be determined which is more feasible. Innovative materials allow higher temperatures and higher burnups. Such fuels and materials become a more realistic possibility when designed using advanced simulation techniques.

The present time is an exciting one for fast reactors. The restart of Monju and the entry into service of the CEFR, PFBR and BN-800 are on the horizon. Operation of these reactors will bring new data and knowledge, in addition to possibilities for international collaboration. New strategic requirements for the fast reactor mission are being introduced, such as waste management and high breeding, and regional approaches are emerging. It can be anticipated that R&D dominate for the next 10–20 years, guided by imaginative breakthrough to cope with the most crucial issues and provide a clear focus on the objectives.

The International Conference on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09) was organized by the IAEA as a timely response to strong Member States' demand and whose expectations were met by the large gathering of experts in attendance. The conference was regarded as a particularly successful meeting, with positive remarks and appreciation given to its format, the quality of submissions and the stimulating discussions which took place both inside and outside of the official sessions.

Moreover, the success of the conference itself marked a good start for the revival, following an 18 year hiatus, of periodic conferences dedicated to fast reactors and related fuel cycles, for which it was suggested that such conferences be held in the future at three year intervals and at different venues.

OPENING SESSION

Chairpersons

S. TANAKA
Japan

Yu. SOKOLOV
IAEA

OPENING ADDRESS

T. Okazaki
Japan Atomic Energy Agency,
Ibaraki, Japan

Good morning, ladies and gentlemen.

As Conference General Chair of the International Conference on Fast Reactors and Related Fuel Cycles (FR09) organized by the IAEA, and as a representative of the host organization for this conference, I would like to deliver an opening address.

First of all, I would like to express my appreciation that so many participants, both from home and abroad, are attending this conference. Above all, I'm most grateful for the commitment that the International Advisory Committee, the International Scientific Programme Committee, the Local Organizational Committee and the Local Executive Committee members have shown in holding this year's conference, FR09.

For this conference, about 750 participants have registered from 26 countries and three organizations (European Commission, OECD Nuclear Energy Agency, IAEA). I'm grateful that so many people are very interested in fast reactor development.

Thinking back on the history of the conference of fast reactor systems, it all started back in 1974 in London. It then continued to be held every few years up until the fifth Kyoto conference in 1991. However, it has been suspended since then and so now, this year, the conference is being held for the first time in 18 years at the same location where we left off in 1991, the Kyoto International Conference Center.

During this period, in the early 1990s, the FFTF and EBR-II experimental reactors in the United States of America were shut down. In 1991, the construction of the SNR-300 prototype reactor in Germany was cancelled for both economic and political reasons, and in 1994, the operation of the PFR prototype reactor was stopped in the United Kingdom. Then, in 1998, the Super Phénix demonstration reactor in France was also shut down. In Japan, there was a sodium leak accident at the Monju prototype reactor in 1995 during a plant performance test.

On the other hand, since 2000, the importance of nuclear energy has been recognized once again as a global energy source for the new century. In 2000, the Generation IV International Forum and the IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles were launched as a new framework for multilateral nuclear cooperation. In 2006, the Global Nuclear Energy Partnership started. There has been a new trend in fast reactor development in the nuclear renaissance worldwide.

It has been reported that China's CEFR experimental reactor is nearly reaching criticality. The BN-800 demonstration reactor in the Russian Federation and India's PFBR prototype reactor are in preparation for construction. In Japan, Monju, which has long been suspended, is now being prepared for its restart within this fiscal year, by the end of March 2010, and the FaCT project has been promoted as one of the key national technologies aiming at the commercialization of future sodium cooled fast reactor cycles.

Thus, global fast reactor development has just overcome a period of 'winter-like' hardship and has entered a new stage of commercialization. There are two key phrases to describe the new period of fast reactor development: "stop global warming" and "prevent the threat of nuclear weapons".

Regarding the global warming issue, 12 years ago, that is, in 1997, the Kyoto Protocol was adopted at the Third Conference of the Parties to the United Nations Framework Convention on Climate Change (COP3), which was held in Kyoto. The COP15 was held in Copenhagen with the goal of forming a framework for greenhouse gas reduction after 2013.

We are aiming to achieve the world common target of reducing by half the emission of greenhouse gases by 2050. It is impossible to reach a solution on this issue without a long standing nuclear energy supply. Particularly when considering the recent rapid increase in the price of natural uranium, the necessity for fast reactor development should again be internationally recognized from the viewpoint of achieving significant effective utilization of uranium resources and decreasing the impact on the global environment, thanks to the reduction of radioactive waste.

Regarding prevention of the threat from nuclear weapons, in Prague in April 2009, US President Barack Obama stated the USA's commitment to seek the peace and security of a world without nuclear weapons and garnered worldwide sympathy. Then, in September, resolutions on non-proliferation and nuclear disarmament were proposed at the United Nations Security Council.

While the Nuclear Security Summit and the NPT Review Conference are to be held this spring, Y. Amano assumed the post of Director General of the IAEA, responsible for promoting the peaceful use of nuclear energy and non-proliferation.

Against a mood of growing concern over nuclear arms reduction and non-proliferation moving towards a world without nuclear weapons, the securing of non-proliferation is absolutely crucial for fast reactor system development aimed at prolonged support for the peaceful use of nuclear energy.

The purposes of this conference are to confirm and to discuss significant issues on fast reactor development and related fuel cycle development, as well as to exchange information on achievements of projects and developments multilaterally for its efficient promotion.

Some events to be held include the Young Generation Event to foster discussion among young researchers and students, in addition to oral presentations, poster sessions and panel discussions. Special events such as the Tsuruga session for promoting communication between specialists and the public, as well as the Monju tour on the final day, are also planned.

We hope this conference will lead to a fruitful debate and that international cooperation, and research and development of fast reactors will be further advanced, thereby contributing to knowledge creation and technological development for the prosperity of humankind.

Thank you.

OPENING ADDRESS

Y. Kawabata
Ministry of Education, Culture, Sports, Science and Technology,
Tokyo, Japan

Presented by K. Hakozaki

Good morning, distinguished delegates, ladies and gentlemen. I would like to express my deep gratitude for your presence at the "International Conference on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities" organized by the IAEA. I would like to make a brief opening address on behalf of MEXT[1].

Firstly, I would like to welcome all who have travelled the long distance to Japan, and to express my thanks to people in Japan for their usual acceptance and for their cooperation on the research, development and use of nuclear technology. I would also like to thank the staff of the IAEA, the Japan Atomic Energy Agency and the commissions for their commitment to organizing this meeting.

Today, humankind faces global issues on a scale never before seen, including global warming and energy resource security. Under such circumstances, ensuring the energy supply is essential for solving both the energy problem and global climate change simultaneously. This is increasingly being recognized all over the world.

Sharing the recognition, we promote research, development and the use of nuclear energy as the major source of electrical power. We are aiming at the establishment of the fast breeder reactor cycle, which will ensure a long term energy supply, through efficient use of uranium resources.

At MEXT, we continue to promote research and development in order to achieve the early commercialization of the fast breeder reactor cycle, by utilizing the prototype Monju fast breeder reactor.

We are now doing our utmost to restart Monju by the end of March 2010, with the acceptance and cooperation of the local community. After the restart, we will enhance the reliability of Monju as an operational power plant, drawing upon operational experience. At the same time, we will continue research and development of radioactive waste reduction for topics such as minor actinide burning, as well as the enhancement of nuclear non-proliferation. We hope that Monju will play an important role, not only domestically, but also globally as one

[1] MEXT: Ministry of Education, Culture, Sports, Science and Technology.

of the few high-end fast reactors. This makes Monju a major global centre in the area of nuclear fuel cycle research and development.

In this conference, we can share the fruits of each country's and each organization's research and development for the purpose of realizing the nuclear fuel cycle, which is the important international political theme.

Finally, I would like to express my wishes that this conference contributes to the stimulation of further research and development on the fast reactor cycle through active discussions and, ultimately, brings significant benefits to all countries.

Thank you for your kind attention.

OPENING ADDRESS

Y. Amano
International Atomic Energy Agency,
Vienna

Distinguished guests, ladies and gentlemen.

It is my honour to address participants at this opening session of the International Conference on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities, organized by the IAEA and hosted by the Japan Atomic Energy Agency.

Fast reactor technology has the potential to ensure that energy resources which would last hundreds of years with the technology we are using today will actually last several thousand years. In other words, it can satisfy enormous increases in demand.

This innovative technology also reduces the risk to the environment and helps to limit the burden that will be placed on future generations in the form of waste products.

The coming year will be an exciting one for the development of fast spectrum nuclear reactors. We expect to reach several important milestones:

(a) The first criticality of the China Experimental Fast Reactor;
(b) The restart of the Monju prototype fast reactor in Japan;
(c) The new insights we will gain through the end-of-life studies at the Phénix reactor in France.

In the near future, new fast reactors will be commissioned: the 500 MW(e) Prototype Fast Breeder Reactor in India, the first in a series of five of the same type, and the BN-800 reactor in the Russian Federation. Moreover, China, France, India, Japan and the Republic of Korea are preparing advanced prototypes and demonstration or commercial reactors for the 2020–2030 period.

Nuclear power is set to become an increasingly important part of the global energy mix in the coming decades as demand for energy grows. A number of countries in both the developed and developing world have told the IAEA that they are interested in introducing nuclear power. The 30 countries which already have nuclear power reactors are set to build more.

This trend is likely to be accompanied by accelerated deployment of fast reactors. Continued advances in research and technology development are necessary to ensure improved economics and maintain high safety levels with increased simplification of fast reactors.

The number of countries with fast reactor development programmes is increasing steadily. Emerging economies are joining the traditional fast reactor technology holders and pursuing important research and technology activities.

The IAEA provides a unique collaborative framework to enable all these players to work together to ensure that innovative fast reactor technology progresses. We provide an ‘umbrella’ for knowledge preservation, information exchange and collaborative R&D in order to pool resources and expertise.

Our Technical Working Group on Fast Reactors promotes the exchange of information on national and multinational programmes and new developments and experience. It aims to identify problems, help find solutions and facilitate practical application of fast neutron systems.

In the Programme and Budget Cycle for 2010–2011, IAEA projects on innovative fast neutron systems will continue to focus on issues addressing fast reactor economics, enhanced safety characteristics, sustainability and public acceptance.

As far as public acceptance is concerned, I believe there is a growing understanding throughout the world that clean, efficient and safe nuclear energy has a key role to play in meeting the growing demand for energy while minimizing damage to the environment.

Fast reactor technology has a promising future. The IAEA will continue to work with all of you to help interested Member States to benefit from it and to establish, or further enhance, the necessary safety, security and safeguards infrastructure.

Let me conclude by expressing my gratitude to all of the dedicated colleagues in the International Advisory Committee, the International Scientific Committee and the Local Organizing Committee who have worked hard to organize this conference. I wish you every success in your deliberations over the next few days.

OPENING ADDRESS

JAPAN'S NUCLEAR REACTOR STRATEGY

S. Kondo
Japan Atomic Energy Commission,
Tokyo, Japan

Thank you very much Mr. Chairman for your kind introduction. Distinguished colleagues, ladies and gentlemen, it is a great pleasure for me to have the chance to address you here in Kyoto at this "International Conference on Fast Reactors and Related Fuel Cycles (FR09)". At the outset, I would like to thank the IAEA for organizing this conference and, taking this opportunity, I would like to assure its new Director General, Y. Amano, of Japan's continuing support for the IAEA. I am looking forward to continuing to work with the IAEA in order to extend the benefits of the peaceful uses of nuclear energy and science and technology to a global population.

We are witnessing today a global emergence of interest in the construction of nuclear power plants. There are a number of reasons for this. Major factors are the urgent and ever growing need for energy, particularly in the developing world, fluctuations in fossil fuel prices, the pursuit of security of energy supply and the growing recognition of the need to combat global warming.

Despite the global economic crisis, the IAEA's latest projections continue to show a significant increase in nuclear generating capacity in the medium term. The low projection for 2030 is now 511 GW(e) of generating capacity, compared with 370 GW(e) today. The high projection is 807 GW(e); more than a doubling of present levels.

Most of the 30 countries that already use nuclear power plan to expand their output. Growth targets have been raised significantly in China, India and the Russian Federation. In addition, according to the IAEA, some 50 countries — mostly in the developing world — have informed the IAEA that they might be interested in launching nuclear power programmes and 12 of these are actively considering nuclear power.

Even in the high case projection, however, nuclear power's share of global power generation will go down from the current 16% level to 14% by 2030 and then rise to 22% by 2050, according to the projection published by the OECD Nuclear Energy Agency in 2008[1]. In other words, the growth of nuclear power in

[1] OECD Nuclear Energy Agency: Nuclear Energy Outlook — 2008.

the global power sector will not be able to keep pace with the growth in global electricity demand, at least in the medium term.

Then what should the global nuclear community do before ‘dawn’, preparing for the day when nuclear energy will play the leading role in global energy supply. My answer is, let us promote carefully planned yet highly aggressive actions across three different time frames: short term, medium term and long term.

The major short term action should be to continue to operate existing reactors safely and reliably, maintaining the public’s trust in both plant operators’ safety management and the government’s regulatory activities for safety and security.

In the case of Japan, urgent action in this category is to complete the re-evaluation of the seismic safety of every nuclear facility in Japan, taking into consideration lessons learned from the July 2007 seismic event at the Kashiwazaki-Kariwa nuclear power plant on the propagation of the seismic wave generated in a nearby fault, in which the seismic input to the plant significantly exceeded the level of the design basis seismic input. It is to be hoped that this review for the prototype Monju fast breeder reactor will be completed very soon.

The major medium term actions in the case of Japan are to add new generating capacity steadily, to operate the Rokkasho Reprocessing Plant steadily and to construct intermediate spent fuel storage facilities in a timely manner, to provide assistance to countries that are considering introducing nuclear power to build the necessary infrastructure and to train a young generation of nuclear scientists and engineers who are to sustain the development and utilization of nuclear energy in the future.

One of the major long term actions should be to promote research and development programmes that exploit nuclear energy’s innate feature, namely, its economically harvestable resource base, which is good for a millennium, by closing the nuclear fuel cycles using fast neutron reactors.

You have gathered here in Kyoto to discuss the challenges and opportunities of this programme. I am sure that this city is one of the best places in the world for having discussions about such long term issues, since Kyoto had been the capital of Japan for more than a thousand years and it has a rich and unique cultural approach to long term prosperity.

In the previous century, there were several active fast breeder reactor R&D programmes being pursued worldwide. However, commercial development of fast reactors was put on hold in the 1980s and 1990s for several reasons, but primarily because they were projected as being uncompetitive.

At the start of this century, however, recognizing that the environmental benefits of nuclear energy could even extend to other energy products besides electricity by the latter part of this century, not a few countries have started to consider it wise to promote, as a long term action, a significant R&D effort on fast reactors and closed nuclear fuel cycles that meet the technology goals in sustainability, economics, safety and reliability, proliferation resistance and physical protection that will help nuclear energy play an essential worldwide role in the future.

In the case of Japan, the Japan Atomic Energy Agency (JAEA) is promoting the R&D of fast reactors and fuel cycle technology that can make it competitive and sustainable with regard to the energy supply market beyond 2050. Japan's current programme goals are to produce, by 2015, a conceptual design for a fast reactor and its fuel cycle system that can satisfy the performance goals with regard to safety, economy, sustainability and proliferation resistance, and to start the operation of its demonstration system by around 2030.

Currently, the JAEA is exploring candidate technologies for a sodium cooled fast reactor that loads mixed oxide (MOX) containing minor actinides as constituents. Specifically, it is exploring advanced reprocessing technology that can efficiently recover minor actinides as well as plutonium from spent fuel and developing advanced technology to fabricate such fuel so as to make the fast reactor and its fuel cycle system a very unattractive route for diversion of weapons-usable material.

Furthermore, it has been claimed that selective separation of the various long lived actinides from spent fuel in the reprocessing process would allow their fabrication into fuel or targets to be irradiated in specifically adapted fast reactors or in accelerator driven systems where they would be transmuted into shorter lived elements while contributing to energy production, thereby leading to a reduction in the volume and radiotoxicity of the waste to be disposed of.

The set of technology goals deliberated at the outset of the project has been quite effective in stimulating the search for innovative technology candidates. When we decide a system design is to be taken for further development, it becomes necessary to convert these candidates into a set of decision criteria. As there is a gap between available and required knowledge, to do so involves project risks.

Fast reactors will allow the recycling of used MOX fuel that is not practicable in light water reactors. This is an intrinsic advantage of fast reactors and from the sustainability viewpoint this will make it possible to allow the whole amount of high level waste to be disposed of in the usual form of glass canisters and with similar hear generation characteristics.

The criteria of economic competitiveness comes from the requirement of the market place and it is imperative therefore that the life cycle and power generation costs and financial risk of the system proposed should be at least comparable with those predicted for the light water reactors over the 2050 time frame and based on a specific risk assessment.

As for safety, we have regulatory requirements such as safety goals and even quantitative safety objectives in terms of core melt frequency for light water reactors in some countries. Therefore, they should be used as references for considering the acceptance of the system proposed, though the differences between the core melt phenomena in light water reactors and those in sodium cooled fast reactors and their impact upon the applicability of these requirements should be clarified beforehand.

If not currently an issue, it will become one of the major risk elements for the project in the future. As for nuclear security, a procedure has already been established in many countries, in compliance with the IAEA's INFCIRC 225 (i.e. IAEA Guidelines on the Physical Protection of Nuclear Material and Nuclear Facilities), to define a design basis threat that outlines the set of adverse characteristics for which the facility operators and state organizations together have protection responsibility and accountability.

Unlike the safety area, however, the nuclear community continues to be pressured into increasing security. Obviously, this is because quantitative security risk analysis is still at an early stage and the quantitative security objectives have not yet been established in society. I hope that a more balanced view on this issue will prevail in the near future.

The situation is far vaguer for non-proliferation goals. The obligation under the NPT for its State Party is to put any nuclear facility under IAEA safeguards. In September 2009, the United Nations Security Council resolved to encourage efforts to ensure development of peaceful uses of nuclear energy in a framework that reduces proliferation risk, adhering to the highest international standards for safeguards.

Why is a framework mentioned in this resolution? Presumably it is in recognition that the proliferation concerns should come not from the facilities themselves but from the possible actions to be taken by a country.

One can identify this recognition clearly in a speech given by the former IAEA Director General, M. ElBaradei, made at an IAEA conference in Beijing in 2009. According to him, countries that have mastered uranium enrichment and plutonium separation, much more than those that have mastered sophisticated nuclear fuel cycle technology, such as that required to handle highly radioactive and 'hot' materials such as minor actinide-bearing liquid, can be viewed as nuclear weapon capable States, meaning they could develop nuclear weapons within a short time span if they left the NPT or launched clandestine programmes[2]. He claims that the NPT gives too narrow a margin of non-

[2] One school of thought claims that the probability of failure to make nuclear weapons covertly and overtly after leaving the NPT should be made sufficiently high by eliminating any kind of nuclear fuel cycle facilities, so as to give the United Nations Security Council ample time to consider any intervention.

proliferation and therefore a multinational approach to the entire fuel cycle, including the back end, has great potential to facilitate the expanded safe and secure use of nuclear energy for peaceful purposes, while reducing the risk of proliferation.

It should also be mentioned that a series of recent G8 summits has asked the Nuclear Suppliers Group to establish guidelines to restrict the transfer of reprocessing technology, which is an essential element for the utilization of fast reactors.

Considering these developments, the global fast reactor R&D community should ask themselves, from the viewpoint of project risk management, whether they should pursue the development of fast reactor systems that fit into a global society with a large scale regional fuel cycle centre under multilateral control. There is a proposal that this centre will provide a cradle to grave service to operators of fast reactors, supplying fresh fuel that contains fissile plutonium just in time for loading to the reactors and immediately taking back the used fuel when it is removed from the reactors. In such cases, it might be unnecessary to recycle minor actinides as it can be claimed that to do so has no particular advantage in terms of safety of high level waste disposal and minor actinide-bearing fuels feature a potentially considerable increase in gamma and neutron dose and of decay heat, which would require specific protection and cooling methods for transporting these fuels.

As safeguards represent a cog in the wheel of the non-proliferation policy, we run the risk of obtaining an insufficient answer to the request to develop a civil nuclear energy framework if we concentrate our attention only on the means of strengthening the proliferation resistance of the technological systems concerned.

In conclusion, ladies and gentlemen, many countries are committed to making a long term investment in the development of the fourth generation nuclear reactor systems and fast reactors and its fuel cycle, in particular with entrepreneurial imagination and willingness. The key to this endeavour will be to create and deploy new products and processes or new systems that currently do not exist. The energy technologies that catalyse such development will reap the greatest rewards.

Product innovation is, however, not so easy to achieve successfully. Innovative learning is necessary to be successful in this endeavour. I have touched upon an aspect of risk management in this innovation process for a fast reactor and its fuel cycle technology, as I believe risk management is an operationalization of innovative learning.

I found in the programme for this week, many sessions for discussion of innovative learning in various contexts and areas with a view to pursuing the innovation of the fast reactor and its fuel cycle technology. I sincerely wish you every success with the conference. Thank you for your attention.

OPENING ADDRESS

SAFETY OF FAST REACTORS: THE REGULATOR'S APPROACH

M.-P. Comets
Autorité de sûreté nucléaire,
France

1. THE FRENCH NUCLEAR SAFETY AUTHORITY (ASN)

ASN is in charge of the regulation of nuclear safety and radiation protection for:

(a) Around 120 large nuclear facilities;
(b) Several tens of thousands of facilities and activities using sources of ionizing radiation for medical, industrial and research purposes;
(c) Several hundred thousand transports of radioactive material.

ASN is not in charge of the regulation of defence or security but has a role in informing the public about nuclear safety. ASN is independent from the Government; it reports on its activities to Parliament.

2. THE FRENCH CONTEXT REGARDING GENERATION IV

Sodium fast reactors (SFRs) have already been operated in France, in particular SuperPhenix, which was shut down in 1998, and Phenix, which is performing its final tests.

Following the 13 July 2005 Act fixing the guidelines of the energy policy and the 28 July 2006 Act dealing with the sustainable management of radioactive material and waste, research and studies on the new generations of nuclear reactors are to be conducted in order that:

(a) An assessment can be made in 2012 of the industrial prospects of these reactor types;
(b) A prototype installation can be set in operation before 31 December 2020.

The industrial phase is foreseen for 2040–2050.

The choice was made by the French Government to work on two designs: the SFR and the GFR. The French industrial organizations, within the above timescale, are working on the SFR design.

ASN set up an internal Generation IV working group in 2008 in order to be able, when the time comes, to define the safety objectives of Generation IV reactors. ASN has also held regular discussions with the industrial organizations carrying out the SFR project in France.

3. ASN POSITION ON GENERATION IV REACTORS

Comparison between the different designs. ASN asked the French industrial organizations to justify the choice of the SFR design from a safety point of view as compared to the other designs. ASN wishes to reflect upon the safety prospects that can be displayed by the other designs. This consideration has to include the extent of the R&D needed, taking into account the scientific and technical issues, current knowledge and also the separation and transmutation possibilities.

Safety objectives. For ASN, the safety objectives of the 2020 prototype must be at least as ambitious as the EPR ones. The EPR safety objectives, having been set up in the 1990s, are in need of improvement.

Experience feedback of SFR. ASN asked the industrial organizations to assess the national (Rapsodie, Phenix, SuperPhenix) and international experience feedback of this design. For ASN, this assessment is the starting point in the definition of the safety objectives and options and of the associated research and development orientations.

4. ASN ACTION IN THE FIELD OF FAST REACTOR SAFETY

In June 2009, as previously mentioned, ASN asked the French industrial organizations involved in the SFR project for an assessment of the national and international experience feedback of this design and the justification for choosing this design in terms of safety.

Process to define the safety objectives for the SFR design. Based in part on the assessment of the safety questions raised by the SuperPhenix, Phenix and other SFR reactors in operation around the world, ASN wishes to define an iterative process for the definition of the safety objectives before ASN formally discusses the safety options and to discuss this process with the designers, as was done for the EPR. The authorization decree for the creation of the EPR was issued in April 2009, and the safety objectives were defined by the French and German safety authorities in 1993. The examination of the EPR safety options was completed in 2000, leading to the development of technical guidelines for the design and construction of the next generation of nuclear pressurized water reactors. The assessment of the EPR design of the EPR project was then performed within the framework of these guidelines.

Position taking process. Based in particular on the technical directives and on the SFR experience feedback, ASN will have discussions with the industrial organizations from 2009 to 2014 and reach a decision on the safety option based on the submissions from the industrial organizations expected in 2014. The discussions will cover topics such as in-service inspections, chemical hazards of sodium and terrorism.

ASN expects safety improvements to be made over the existing design, including the EPR, Phenix and SuperPhenix.

5. CONCLUSIONS

As we are talking of an industrial phase (2040–2050), ASN considers it important to be able to have a thorough discussion on the relative safety of the different designs.

Concerning the SFR:

(a) National and international experience feedback is the essential first step of the whole process.
(b) Safety objectives must be at least as ambitious as those for the EPR.
(c) It is important to carry out this work at an international level because it contributes to optimizing resources and harmonizing nuclear safety, which is a major issue for the regulators.

OPENING ADDRESS

MEETING TOMORROW'S ENERGY NEEDS

P.B. Lyons
United States Department of Energy,
Washington, D.C.,
United States of America

I want to thank the IAEA, as well as the international and local organizing committees, for assembling this impressive group of conference attendees to share information and exchange ideas this week. I am honoured to be a part of this distinguished panel. I also want to thank our Japanese colleagues for their wonderful hospitality and for selecting the beautiful city of Kyoto as the venue.

This is an historic time of challenge and opportunity. President Obama is seeking to accelerate our nation's transformation to a low carbon economy. He has set a goal of reducing carbon emissions by 80 per cent by 2050. As we meet here in Kyoto, the site of another historic climate change discussion, President Obama is preparing to attend the United Nations Climate Change Conference in Copenhagen, continuing his Administration's commitment to tackling climate change and building a clean energy economy.

Four months ago, Dr. Warren "Pete" Miller was confirmed by the United States Senate to serve as Assistant Secretary for the Department of Energy's Office of Nuclear Energy. Soon afterwards, I joined Pete as his Principal Deputy. In that short time, Pete and I have come to appreciate fully the enormous task we have before us. Transforming our economy from one reliant on fossil fuels to a low carbon future will take investments in energy efficiency and all forms of low carbon energy technologies, including nuclear energy. Our job is to assure that nuclear technologies can contribute to meeting this aggressive goal for reduced greenhouse gas emissions.

Nuclear energy can contribute to the future energy supply in two basic areas: (i) in its traditional role of generating electricity and (ii) as a source of process heat for industrial, petrochemical and desalination purposes. Within the Office of Nuclear Energy, we are now developing a roadmap that will help us in "meeting tomorrow's energy needs" by addressing these two vital areas. We have established five strategic goals as the foundation on which to base our programmes:

(1) Extend the lifetime of existing reactors;
(2) Enable the building of the next and future generations of nuclear power plants;

(3) Reduce the carbon footprint of the transportation and industrial sectors;
(4) Develop a sustainable nuclear fuel cycle;
(5) Understand and minimize proliferation risks.

The first goal is “extending the lifetime of existing reactors”. If we are planning for an increase in the contribution of nuclear power, it is of utmost importance that the existing reactors continue to operate safely and efficiently. US nuclear power plants have maintained a 30 year record of exceptional safety and performance, achieving capacity factors above 90 per cent. As a result, the owners of almost all of these plants have either successfully obtained or are planning to apply for licence renewals that extend the length of their operating licence from 40 to 60 years.

We have launched a research effort aimed at providing, if possible, the technical basis for operating the existing US fleet beyond 60 years. Investment in long term, high risk, high reward research and development may provide the scientific foundation for plant owners to make investment decisions to prolong the economic lifetime of these valuable national strategic assets.

Our second goal is “enabling new plant construction”. Analyses of the climate change issue by independent organizations show that reducing carbon emissions will require a portfolio of technologies and that nuclear energy must be part of that portfolio. These studies project a need to build between 100 and 200 GW(e) of new nuclear generating capacity in the United States of America over the next 30 years.

We began our Nuclear Power 2010 programme seven years ago as a cost shared partnership involving both industry and government to reduce the financial and regulatory risks associated with building new advanced light water reactors (LWRs). These new designs were developed to further advance operational safety and economics when compared to the currently deployed reactors. The programme has done its part to clear the way for many new plants, with six to eight expected to be built by 2020.

One of the challenges facing our deployment of new plants is the availability and cost of capital. Therefore, progress on loan guarantees is very important. Four new projects are currently under consideration for federal loan guarantees. We are hopeful that the first conditional nuclear loan guarantees will be announced before the end of the year.

The large plants now planned for the first wave of construction won’t be the only options considered by US utilities for operation in the 2020–2050 time frame. There is increasing interest in small modular reactors, which offer potential advantages in the way they are financed, manufactured, constructed and licensed. Some of these designs use LWR technologies, while some utilize fast reactor designs. The lower power of these reactors is also well suited to the grid infrastructure of many developing nations.

Our third goal is to expand the low emission benefits of nuclear power beyond electricity production. Half of our nation's carbon emissions come from the transportation and industrial sectors, but nuclear power has not played a significant role in these sectors. As the transportation sector begins to use more plug-in hybrids and electric vehicles, nuclear power can help meet the additional demand for low carbon electricity production.

In addition, harnessing nuclear power as a heat resource for industrial processes could enable nuclear power to increase significantly its current 8% share of our total energy supply. Our goal is to generate low carbon, nuclear driven, process heat for industrial use. Perhaps we can use nuclear power as the heat source to produce unconventional transportation fuels cleanly from our domestic fossil resources or hydrogen to enable more effective use of biofuels.

Our fourth goal, the one that is most closely related to the purpose of this conference, is "developing a sustainable fuel cycle". To support a large expansion of nuclear energy, we must develop a fuel cycle that is economic, safe, secure and environmentally friendly.

President Obama and Secretary Chu believe we can provide better technical solutions than our current once-through fuel cycle. As a result, we're working on a targeted research and development programme studying the back end of the fuel cycle to improve significantly the management of used nuclear fuel. We have refocused this effort on science based, goal oriented research and development that integrates theory, experiment and high performance modelling and simulation to explore alternative fuel cycles and game changing technologies that may produce less used nuclear fuel and lower the long lived actinide content of the final wastes.

We have made great strides in our understanding of various approaches to processing used fuel. But further advances in technology may require that we change the way we think about nuclear material. We will be supporting innovation and creativity to maximize the energy we extract from new fuel. Using safe and secure dry cask storage, we have time to explore various options and arrive at a decision.

Our plan is to perform research and development and develop technologies to demonstrate the best approach in each of three back end strategies:

(1) The first strategy will involve research on the once-through fuel cycle to understand the limits of increasing burnup, both in LWRs and in new reactor types, and the performance of that spent fuel in different geological media.

(2) The second strategy for a modified open cycle would involve some limited processing of used fuel to enable the production of ultra-high burnup fuels that would be disposed of after irradiation. Fast reactor technology will probably be a part of this strategy.

(3) The final strategy is a full recycle approach with extensive processing to remove some elements from the used fuel, reuse some of them in fast reactors, possibly transmute others and minimize the volume and toxicity of the final waste products.

These first two strategies are likely to involve fast reactor technology. With this technology, nearly 58 years ago on 20 December 1951, the Experimental Breeder Reactor (EBR-I) provided the first useful electricity from nuclear energy, powering four light bulbs in the Idaho desert. This provided the first evidence of the enormous potential for fast reactor technology to satisfy future energy needs. However, with the shutdown of the EBR-II in 1994, the USA is without an operating fast reactor. Our infrastructure to support fast reactor development has continued to deteriorate.

Although we have shifted our efforts in the fast reactor area away from accelerated commercial deployment, fast reactor technologies clearly have promise. We will continue to explore innovative technologies for fast reactors to reduce costs, improve performance, enhance safety and better manage nuclear waste and fuel resources.

Our fuel cycle research and development efforts will be guided by the goal of enabling a national decision to deploy a complete waste management system by 2050. This time frame allows us to advance the state of the art in each of these strategies and demonstrate technologies by 2030. Of course, no matter what our research and development produces, we cannot forget that the nation must ultimately have at least one geological repository.

Our final imperative is related to non-proliferation, “to understand and minimize proliferation risks”. Internationally, there is a surge of interest in nuclear power. Many countries with no previous experience with nuclear power are planning to establish civilian nuclear power programmes. The international community requires that control systems be in place to prevent proliferation and other security threats.

Former IAEA Director General M. ElBaradei provided global leadership on approaches to assured fuel supplies that avoid the need for many countries to construct enrichment or reprocessing facilities. We can ensure that nations have access to fresh nuclear fuel through the use of multilateral fuel supply assurances, international fuel banks and multinational used fuel ‘take back’ strategies.

This approach was confirmed at the Global Nuclear Energy Partnership Ministerial Meeting in Beijing in October 2009, where the Executive Committee agreed to “explore ways to enhance the international framework for civil nuclear energy cooperation, including assurances of fuel supply, so that countries can access peaceful nuclear power without increasing the risks of proliferation. Cradle-to-grave nuclear fuel management could be one important element of this framework.” Such a framework is a priority of the USA and will be critical as we simultaneously address the challenges of climate change, energy security and non-proliferation.

To support the investigations required to meet these five imperatives, the Department of Energy retains expertise and capabilities specifically suited to nuclear energy research, development and demonstration. By working with industry, involving our nation’s universities and cooperating with international organizations, we leverage capabilities, share facilities and more effectively advance technology development. As we continue to pursue advanced fuel cycle technologies, international collaborations are a must. Nuclear technology development and demonstrations are too costly for a single nation to fund alone.

In conclusion, meeting tomorrow’s energy needs, both in the USA and around the globe, presents both challenges and opportunities. The Department of Energy has defined an agenda to support the effective deployment of safe, clean and secure nuclear energy, both domestically and internationally, to help secure tomorrow’s global energy needs.

OPENING ADDRESS

FAST NEUTRON REACTORS AND SUSTAINABLE DEVELOPMENT

J. Bouchard
Commissariat à l'énergie atomique,
Gif-sur-Yvette, France

The aim of this presentation is to provide an insight into the challenges that lie ahead for the development of fast reactors.

From the moment when the first fast reactor — EBR1 — lit up the city of Arco right up to Superphenix, by far the largest fast reactor ever built, there have been 40 years of fast reactor development, mainly centred on sodium cooled systems, leading to the successful operation of such plants. Therefore, the question could arise about the need for more R&D and the relevance of new prototype designs.

There have been two major development steps in the history of fast reactors. During the 1960s and 1970s, their development was undertaken following concerns related to the energy supply, resulting mainly from the oil crisis, as well as from the need to use uranium resources more efficiently. In the 1980s, however, demand for nuclear energy declined after the Three Mile Island and Chernobyl accidents, as well as from the belief that fossil energy was plentiful and would remain cheap. It took about 20 years to realize that nuclear energy would expand, owing to the energy and climate challenges the world was faced with, and with that, the need for fast reactors became obvious in order to account for the constraints of such expansion.

Currently, however, the context has changed since the 1970s, and the development of fast reactors needs to be made on a new basis, taking into account new criteria linked to economy, safety, reliability, resource saving, waste minimization and physical protection against terrorism or proliferation. Such huge technological challenges also require that the new fast reactor designs be developed internationally, within multinational cooperation frameworks. Such is the goal of the Generation IV International Forum (GIF), which is a gathering of the major key actors in the field of R&D, cooperating for the sustainable development of nuclear energy.

A new way of thinking has emerged from this new context: the awareness that a global solution is required, accounting not only for fast reactors and their associated fuel recycling, but also for full burning of actinides created in both light water reactors and fast reactors. This will provide a solution for resource scarcity as well as for waste and proliferation concerns.

Another important point is the economy. The European Fast Reactor project elaborated at the end of the 1990s is still the most advanced design for a large breeder. It incorporates feedback experience from the construction and operation of Superphenix, as well as various improvements aimed at cost reduction. Subsequent studies showed that though economic competitiveness with third generation advanced light water reactors for electricity production could be attained with the EFR, such a result would require larger investments, making capital cost a major obstacle to the development of reactors based on such a design. Hence, improvements on the economy of fast reactors are today mainly focused on reduction of capital cost.

It is one of the reasons why discussions on the technological choices have been reopened. The first topic is the coolant, which is the major technology driving force for reactors, in particular for fast reactors that cannot use the simplest one, namely water. Sodium was unanimously chosen by scientists and engineers in the 1960s due to its numerous advantages (high conductivity, low viscosity, compatibility with steels, low cost). It has, however, shown its limitations (reactivity with air, opacity). Consequently, the debate has been reopened, especially within the GIF, on other possible coolant options. Moreover, in the event of a large expansion in the use of fast reactors resulting from a wide increase in nuclear energy demand worldwide, it is not mandatory, and even may not be reasonable, that all these fast reactors rely only on sodium technology.

Lead as a coolant is another option being studied by numerous laboratories, mainly in Europe, because of its ‘quiet’ behaviour with air and water. Gas is also considered to be an alternative to sodium in France and a longer term challenge as it could allow for high temperatures and thus possible use for industrial applications other than electricity production.

Of all the different coolant technologies, not one can be singled out as the best one, because each one of them has its own advantages and drawbacks. Though there is common agreement that sodium technology is already mastered, one should always keep in mind that alternative or backup solutions are required.

Two basic designs have been developed for sodium cooled fast reactors: (i) the pool (or integrated) and (ii) the loop concepts. The former one has been adopted for Phenix and Superphenix: operation of these plants was successful and allowed for extensive feedback experience for the fuel as well as for the technology, especially in the case of Phenix. Demonstrations were made, in particular in the field of actinide burning. Most difficulties encountered during the Phenix operation were overcome over a relatively short period of time and without too much difficulty, such as small sodium fires and problems with steam generators and with intermediate heat exchangers. Most difficulties encountered mainly stemmed from minor problems related to materials or design, but none related to the concept itself.

In addition to the lessons learned from Phenix and Superphenix, findings were also obtained from the satisfactory operation of the Russian BN600, which has been running for 30 years now. Other reactors are being built around the world based on the same integrated concept, namely, the CEFR in China, PFBR in India and BN800 in the Russian Federation.

Nevertheless, besides these projects, which will allow us to gain greater experience on the integrated concept, we still have to find ways to reduce the capital cost. Innovative solutions are thus being investigated, which could also improve the features of such reactors, especially their safety. In particular, new options are being looked into for the energy conversion system (gas or supercritical CO_2).

The loop concept is mainly studied in Japan, as the work performed in Germany was interrupted some 20 years ago. The Monju prototype should restart soon and its comeback is awaited by the international fast reactor community as an important tool for future development. In parallel, works are ongoing on the JSFR project to explore all the possibilities offered by the loop concept.

When comparing both concepts, loop and pool, the same conclusion can be drawn as when comparing different coolants. Neither one of them can be considered as the best concept, each one offering certain specific benefits but each also having some drawbacks. Selection should not be made prematurely and investigations of several concepts should be carried out, in order to offer alternative solutions.

The same logic has prevailed since the 1960s between the boiling and pressurized water reactors, thus providing the utilities in particular with freedom of choice. In the loop and pool concepts, investigations should be continued, with a common goal of reducing capital costs, enhancing safety, etc. Some of the developments performed today are common to these different design choices, in particular those related to the fuel.

Various possible fuels are being investigated, namely metal, carbide, nitride and oxide. Most of the past experience has been accumulated on oxide, which is still the reference choice for existing reactors or those that are under construction. Metal fuel was developed in the United States of America (EBR2). Each type of fuel has intrinsic advantages. Carbide and nitride enable the combination of high breeding gain and large margins with respect to melting (gain in performances or in safety), but they are more risky to handle (pyrophosphoricity). Feedback experience may lead to a preferential choice, most probably oxide for near term development, but various options should remain open. Progress has been made for concepts using metallic cladding for the fuel (cladding with no swelling), which seem to point to oxide dispersion strengthened steel as the best solution for the future.

On the safety front, numerous studies are being carried out on the basis of the new requirements mentioned earlier. For example, a strategy has been set up to deal with severe accident management. Provisions are being made for mitigating the core melting risk and, in the event of a core meltdown, for preventing high energy density accident sequences.

Most of the works mentioned so far have concerned sodium cooled fast reactors. As regards concerns gas cooled fast reactors, seen as an alternative to sodium cooled fast reactors, one goal sought is to obtain both the advantages of such reactors for the fuel cycle as well as the advantages of high temperature reactors for applications other than electricity production. This is an important challenge and all the work performed so far has mainly been limited to paper studies; hence the need to build the first experimental gas cooled fast reactor of limited power, around 50 MW(th), to test the technology solutions. A project to build such an experimental gas fast reactor is currently being examined in Europe, and a decision should be made by 2012. The choice of fuel for the gas cooled fast reactor is the major difference with liquid metal coolant systems, mainly due to temperatures but also because of the volume occupied by the gas, which requires a compact fuel with high resistance. Also in the case of the gas cooled fast reactor, safety studies will be of utmost importance, and extensive work is already being carried out, such as, for example, the analysis of gas cooled fast reactor fast depressurization accidents.

Another fast reactor studied is the lead cooled fast reactor. More basic developments are still required, as no final choice has been made as to which coolant would be the best one between pure lead and lead–bismuth alloy. The goal is to obtain a coolant without the same chemical risk posed by sodium. However, lead poses other problems (corrosion, need for operation at higher temperatures).

Various prototype construction projects exist round the world. Among them, let us mention the French sodium cooled fast reactor prototype project, ASTRID. It is a fourth generation prototype scheduled to be operational by 2020. Technological studies are being carried out and a decision to pursue this project will be made in 2012.

In conclusion, it is important to recall the present context, which requires international cooperation for R&D as well as for prototype/experimental reactor construction. Though there exist several national prototype construction projects, harmonization is of great importance in order to avoid duplication and seek complementarities. Pooling of efforts and sharing of R&D tools and construction capabilities will allow for optimization of means.

In the safety and security (non-proliferation and physical protection) areas, it is important to establish international standards, owing to the fact that these matters are largely congruent among the international community. This will allow for the establishment of reference regulatory practices and regulations, as well as international consensus on common (or compatible) high level safety and security objectives.

NATIONAL AND INTERNATIONAL FAST REACTOR PROGRAMMES

(Plenary Session 1)

Chairpersons

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FAST REACTOR DEVELOPMENT FOR A SUSTAINABLE NUCLEAR ENERGY SUPPLY IN CHINA

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Abstract

Nuclear energy is a new member of the energy supply family in China. Satisfactory operating records of all 11 nuclear power plants in China encourage its stepwise and large scale use and the PWR–FBR route matched with a closed nuclear fuel cycle forms a basic strategy. The sufficient utilization of nuclear resources and the treatment of highly radioactive waste by transmutation in fast reactors are the key issues for a sustainable development of nuclear energy. As the first step in FBR engineering development, the 65 MW(th) China Experimental Fast Reactor is approaching startup, the conceptual design of the 600–900 MW(e) China Demonstration Fast Reactor (CDFR) has been started and the 1000–1500 MW(e) China Demonstration Fast Breeder Reactor is under consideration. Three FBR development strategy targets are as follows:

(1) To start realizing CDFR type commercial utilization in small batches by 2030;
(2) To increase nuclear capacity to 240–250 GW(e), representing about 16%, mainly through FBRs by 2050;
(3) To replace coal fired plants by nuclear power on a large scale in the period 2050–2100.

1. INTRODUCTION

China needs a huge energy supply to support its national economic development and improvement in the living standards of its 1.3 billion population. Nuclear energy is a new member of the energy supply family in China. As shown in Table 1, a satisfactory operating record of all 11 units of nuclear power plants, especially with the total average load factor of 85.8% for all nuclear power plants in the 67 reactor-years since commercial operation of each unit has encouraged the public to believe that nuclear power is a safe, reliable, economically acceptable and CO_2 avoidable energy source which could be used as a base load electricity supplier on a large scale. The Government decided in 2006 to accelerate nuclear power development with the target of 40 GW(e) in operation and 18 GW(e) in construction by 2020. Currently, 19 units with a total

TABLE 1. OPERATING LOAD FACTOR OF NUCLEAR POWER PLANTS IN MAINLAND CHINA

Site	Capacity (MW/type)	Grid date	Load factor (%)								
			2000	2001	2002	2003	2004	2005	2006	2007	2008
Qinshan I	300/PWR	1991-12-15	77.2	94.1	66.9	88.6	99.8	86.72	91.44	81.62	96.39
Daya Bay 1	900/PWR	1993-08-31	85.2	84.9	89.6	89.6	87.2	99.79	80.31	90.85	99.60
2	900/PWR	1994-02-07	84.9	89.1	81.6	84.5	73.6	79.44	99.68	88.29	86.39
Qinshan II-1	600/PWR	2002-02-01			74.9	81.0	82.2	92.76	55.20	65.69	87.38
II-2	600/PWR	2004-03-11						85.19	90.30	90.70	86.48
Lingao 1	984/PWR	2002-04-05			92.0	76.8	87.76	82.69	89.16	86.25	90.79
2	984/PWR	2002-12-15				85.0	79.9	90.57	91.89	87.31	84.56
Qinshan III-1	700/PHWR	2002-11-10				90.2	77.3	84.05	98.20	88.35	93.48
III-2	700/PHWR	2003-06-12				90.4	94.0	81.05	88.70	99.87	89.34
Tianwan 1	1000/PWR	2006-06								65.59	74.43
2	1000/PWR	2006-12								78.76	85.50
Total capacity	8.6 GW(e)	Average load factor					85.8%				

capacity of about 20 GW(e) are under construction and another 7 units of total capacity 7 GW(e) have been approved by the Government and the preparations for their construction are under way.

2. FAST REACTOR DEVELOPMENT STRATEGY STUDY

In 1991, it was envisaged after analysis of various domestic energy resource supplies by the 863 High Tech Programme (initiated in 1986), and confirmed in 2005–2007 by the China Engineering Academy and the China Science Academy, that the nuclear power capacity should reach 240–250 GW(e) by 2050 (Table 2).

For the sustainable supply of nuclear energy, as the principal strategy, the PWR–FBR route matched with a closed nuclear fuel cycle has been decided by the Government for a long time. The suggested FBR development strategy and the electrical power development scenario are shown in Table 3 and Fig. 1, respectively, and the realization of three FBR development strategy targets is suggested as follows:

(1) To start realizing the China Demonstration Fast Reactor (CDFR) type commercial utilization in small batches by 2030;
(2) To increase nuclear capacity to 240–250 GW(e), representing about 16%, mainly through FBRs by 2050;
(3) To replace coal fired plants by nuclear power on a large scale in the period 2050–2100.

TABLE 2. ENVISAGED PRIMARY ENERGY PRODUCTION IN CHINA FOR 2050

	1991 envisaged			2005–2007 envisaged	
Energy resources	Exploitable by 2050	Standard coal equivalent (10^9 tsce)	Total requirement (10^9 tsce)	Standard coal equivalent (10^9 tsce)	Total requirement (10^9 tsce)
Oil	0.1×10^9 t	0.45		0.5	
Gas	1500×10^9 m^3			0.3	
Hydraulic	260 ~ 370 GW(e)	0.65		0.6	
Coal	3.4×10^9 t	2.50		2.5	
Nuclear	240 GW(e)	0.60		0.6	
Others		0.30		0.5	
Total		4.5	4.5	5.0	5.0

Note: tsce = tonnes standard coal equivalent.

TABLE 3. SUGGESTED FBR DEVELOPMENT STRATEGY IN CHINA

Reactor	Electrical power (MW)	Design start	Commissioning
CEFR[a]	20	1990	2010
CDFR[b]	600–900	2007	2018–2020
CCFR[c]	n × (600–900)	2015	2030
CDFBR[d]	1000–1500	2015	2028
CCFBR[e]	n × (1000–1500)	2018	2030–2032

[a] CEFR: China Experimental Fast Reactor.
[b] CDFR: China Demonstration Fast Reactor.
[c] CCFR: China Commercial Fast Reactor.
[d] CDFBR: China Demonstration Fast Breeder Reactor.
[e] CCFBR: China Commerical Fast Breeder Reactor.

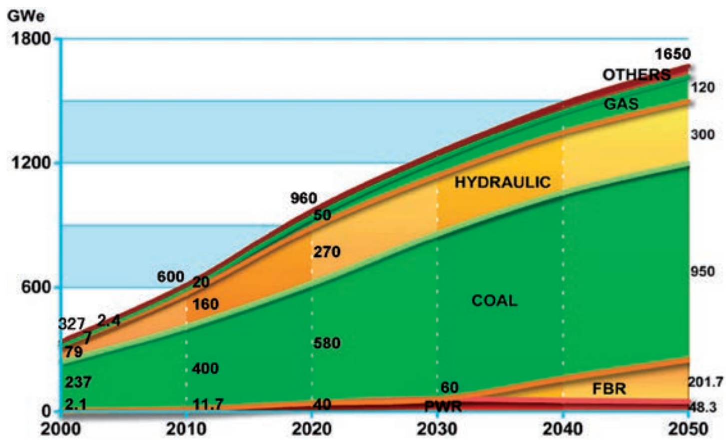

FIG. 1. Envisaged electricity capacity development in China.

Another key issue for the sustainable supply of nuclear energy is to decrease the volume of minor actinides and long lived fission products needing to be buried geologically by burning and transmutation in fast burner reactors. When the CDFR type reactor construction has been realized practically, they could be used as fast burner reactors if the transmutation technology has matured.

The 65 MW(th) CEFR is a sodium cooled pool type reactor (SFR). Its pre-conceptual design was started in 1990 and the first concrete was poured in 2000. Currently, it is undergoing commissioning tests. Experience in the design, fabrication and construction of this type of reactor has been gained.

The possibility of 'jumping' from the 20 MW(e) CEFR up to the 600–900 MW(e) CDFR has been studied. The main benefits are as follows:

(a) The main technical scheme options of design for the CEFR, CDFR and CDFBR have a basic consistency, as shown in Table 4.
(b) A set of computer codes, data files and design criteria of different specialities applied to SFRs have been developed and their validation and verification are under way following the CEFR commissioning and pre-operation testing.
(c) A set of fabrication enterprises and factories having a fabricating licence for nuclear safety grade components for fast reactors have been organized through CEFR construction.

TABLE 4. TECHNICAL CONTINUITY OF CHINESE FBRs

	CEFR	CDFR	CDFBR/CCFBR	CCFR
Power (MW(e))	20	600–900	1000~1500	n × 600–900
Coolant	Na	Na	Na	Na
Type	Pool	Pool	Pool	Pool
Fuel	UO_2 MOX	MOX Metal	Metal	MOX/metal MOX/metal, minor actinides
Cladding	Cr–Ni	Cr–Ni, ODS	Cr–Ni, ODS	Cr–Ni, ODS
Core outlet temperature (°C)	530	500~550	500	500~550
Linear power (W/cm)	430	450~480	450	450~480
Burnup ($MW \cdot d \cdot kg^{-1}$)	60~100	100~120	120~150	100~120
Fuel handling	DRPs/SMHM	DRPs/SMHM	DRPs/SMHM	DRPs/SMHM
Spent fuel storage	IVPS/WPSS	IVPS/WPSS	IVPS/WPSS	IVPS/WPSS
Safety	ASDS PDHRS	ASDS+PSDS PDHRS	ASDS+PSDS PDHRS	ASDS+PSDS PDHRS

Note:
DRPs: double rotating plugs;
SMHM: straight moving handling machine;
IVPS: in-vessel preliminary storage;
ASDS: active shutdown system;
PSDS: passive shutdown system;
PDHRS: passive decay heat removal system.

(d) A good international cooperation environment has been established. However, some R&D and demonstration of key systems and equipment for their reliability have to be conducted and the above-mentioned present favourable bases still need to be advanced.

3. THE CEFR

On the basis of basic research on fast reactor technology (1968–1987) and applied basic research (1988–1993), taking an experimental fast reactor as a target, it was decided to design and construct an experimental fast reactor with a power of 65 MW(th) matched with a 25 MW(e) sized turbine generator — the CEFR. The purposes of the CEFR are:

(a) To incorporate the engineering experience gained on fast reactor design, construction and operation;
(b) As a fast neutron facility, to irradiate and develop fuels and materials;
(c) As a fast reactor overall parameters platform, to test and demonstrate the prototype equipment of fast reactors.

The CEFR is an SFR with $(Pu,U)O_2$ as fuel, but with UO_2 as the first loading, Cr–Ni austenitic stainless steel as fuel cladding and reactor block structure material, bottom supported pool type, two main pumps and two loops for the primary and secondary circuits, respectively. The water–steam tertiary circuit also comprises two loops but the superheated steam is incorporated into one pipe which is connected to a turbine. The general timetable is as follows:

Conceptual design	1990–July 1992
Consultation with Russian FBR Association and optimization	1993
Technical co-design with Russian FBR Association	1994–1995
R&D cooperation with CEA, France	1995–present
Preliminary design	1996–1997
Detail design	1998–2003
Preliminary safety analysis report review	May 1998–May 2000

Architecture construction (first concrete poured) started	May 2000
Reactor building construction completion	Aug. 2002
Installation	2004–2007
Pre-operation testing	2006–2010
Sodium loading of systems	May–June 2009
First criticality	(Dec. 2009)
Connect to the grid	(June 2010)
Full power	(Dec. 2010)

The reactor core, as shown in Fig. 2, is composed of 81 fuel subassemblies. Three safety subassemblies, three compensation subassemblies and two regulation subassemblies, then 336 stainless steel reflector subassemblies and 230 shielding subassemblies, and in addition, 56 positions for primary storage of spent fuel subassemblies are included.

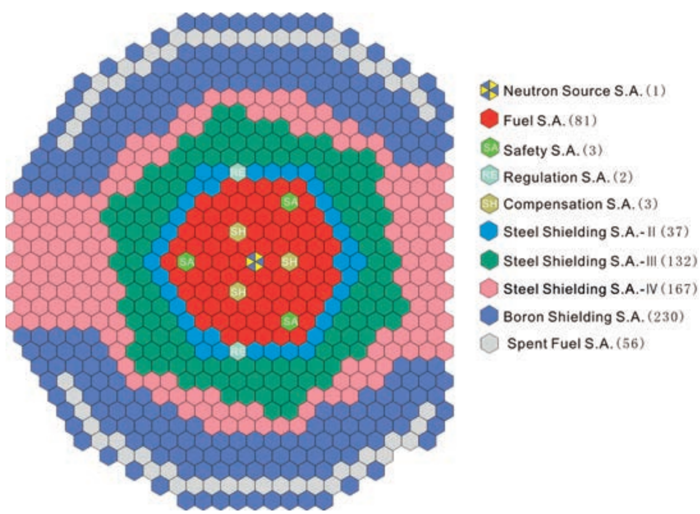

FIG. 2. The CEFR core.

The CEFR block is composed of a main vessel and guard vessel supported from the bottom on the floor of the reactor pit of 10 m diameter and 12 m depth. The reactor core and its support structure are supported on lower internal structures. Two main pumps and four intermediate heat exchangers are supported on upper internal structures. These two structures sit on the main vessel. Two Na–Na decay heat removal system heat exchangers connected to two Na–air coolers which comprise the independent accident decay heat removal system are hung from the shoulder of the main vessel. The double rotation plugs on which 8 control rod driving mechanisms, the fuel handling machine and some instrumentation structures are supported sit on the neck of the main vessel. The main design parameters are presented in Table 5.

TABLE 5. THE CEFR MAIN DESIGN PARAMETERS

Parameter	Unit	Value	Parameter	Unit	Value
Thermal power	MW	65	Diameter of main vessel (outside)	m	8.010
Electrical power (net)	MW	20	Primary circuit		
Reactor core			Number of primary pump		2
Height	cm	45.0	Quantity of sodium	t	260
Diameter equivalent	cm	60.0	Flow rate, total	t/h	1328.4
Fuel $(Pu,U)O_2$			Number of intermediate heat exchangers per loop		2
Pu (total)	kg	150.3	Secondary circuit		
Pu-239	kg	97.7	Number of loops		2
U-235 (enrichment)	kg	42.6 (19.6%)	Quantity of sodium	t	48.2
Linear power maximum	W/cm	430	Flow rate	t/h	986.4
Neutron flux	$n \cdot cm^{-2} \cdot s$	3.7×10^{15}	Tertiary circuit		
Burnup, target maximum	$MW \cdot d \cdot t^{-1}$	100 000	Steam temperature	°C	480
Burnup, first load maximum	$MW \cdot d \cdot t^{-1}$	60 000	Steam pressure	MPa	14
Inlet temperature of the core	°C	360	Flow rate	t/h	96.2
Outlet temperature of the core	°C	530	Plant life	a	30

4. THE CDFR

After pre-conceptual design of the 600 MW(e) fast reactor core, the design study for the CDFR started in 2007 for a capacity of 800 MW(e). Besides the main technical selections, which are settled as indicated in Table 4, some preliminary design boundary conditions are also given in Table 6. The main purpose of the CDFR is to have a commercial, operational power plant and to gain engineering experience with SFRs on an industrial scale. The main demands to this reactor are as follows:

(a) The safety properties should fulfil the recommendations for SFR design given in IAEA-TECDOC-1083[1] and summarized in Table 6;
(b) The reliability should meet commercial nuclear power plant target;
(c) The economics should be acceptable.

A site for it has been selected in Fujian Province.

5. NUCLEAR FUEL CYCLE

To realize step by step the overall targets, the uranium resources should be sufficiently utilized, including by-products plutonium and minor actinides, and the volume of highly radioactive wastes to be geologically buried should be as little as possible. China has mastered the front end technology for PWR fuel supply.

A 100 t/a fuel reprocessing pilot for PWRs and a 0.5 t/a MOX fuel laboratory for CEFR are in the 'hot' testing stage, and a 1000 t/a reprocessing plant and a 50 t/a MOX fuel plant are being designed and planned. China's future energy needs demand a very large nuclear capacity. Consequently, the CDFBR, as the third step in fast reactor engineering development, will use metal fuel, on which R&D has recently been restarted after a break of almost 20 years.

6. SUMMARY

In China, the first phase of nuclear energy application was achieved with the fairly rapid development of PWRs. The second phase, i.e. fast reactor

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Liquid Metal Cooled Fast Reactor Technology, IAEA-TECDOC-1083, IAEA, Vienna (1999).

TABLE 6. DESIGN BOUNDARY CONDITIONS FOR THE CDFR

Parameter	Unit	Value	Parameter	Unit	Value
Power	MW(e)	800	Safety requirements		
Fuel (PuO_2–UO_2)			Reactor core molten probability		$<10^{-6}$/a
Outlet temperature of primary sodium	°C	550	Dose limit at the site boundary not requiring short term off-site response		
Linear power	W/cm	450	Frequency of loss of shutdown function		$<10^{-7}$/a
Breeding ratio		~1.1	Frequency of loss of decay heat removal function		$<10^{-7}$/a
Burnup target (maximum)	$MW\cdot d\cdot kg^{-1}$ heavy metal	120	Load factor	%	>80
First loading (maximum)	$MW\cdot d\cdot kg^{-1}$ heavy metal	100	Reactor life	a	40
Mean duration of reactor run	dMSK-64	300			
Seismic intensity (design)		7			

development, is still at an experimental stage, but three strategic targets for fast reactor development have been proposed to keep nuclear energy safe, ensure a sustainable supply and produce a clean environment.

With these ambitious nuclear targets, the experience gained domestically is obviously not enough. China is willing to have cooperation with IAEA and other countries to share experience and to speed up national nuclear power development.

FRENCH R&D PROGRAMME ON THE SFR AND THE ASTRID PROTOTYPE
(Summary)

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1. INTRODUCTION

Current fast reactor R&D in France is organized around two main topics: (i) the sodium cooled fast reactor (SFR) as the reference option, and (ii) the gas cooled fast reactor (GFR) as the long term option. ASTRID (the Advanced Sodium Technological Reactor for Industrial Demonstration) is the industrial prototype that is foreseen as coming into operation arount 2020. ALLEGRO is an experimental reactor that has been developed at the European level to demonstrate the initial feasibility of a GFR. The topic is focused on the current French SFR R&D programme and plans for the ASTRID prototype.

2. SFR R&D PROGRAMME

The R&D performed in France on the SFR is done in close collaboration between the Commissariat à l'énergie atomique (CEA) and its industrial partners, AREVA and Électricité de France. The R&D programme comprises research in four domains of innovation:

(1) The development of an attractive and safe core, taking into account the specificities of the fast neutrons and sodium, and also the capablity to transmute minor actinides;
(2) A better resistance to severe accidents and external hazards;

(3) The search for an optimized energy conversion system reducing the sodium risks;
(4) The re-examination of the reactor and components' design to improve the conditions of operation and economic competitiveness.

Between 2007 and 2009, the R&D programme provided very useful results and valuable status reports were issued on the followings topics:

- Loop designs;
- Pool designs;
- Review of innovative options such as advanced energy conversion systems, advanced pool/loop designs;
- Fuel handling;
- Impact of reactor power level on safety and costs;
- Core and fuel;
- Safety and severe accidents;
- Status on 9Cr potential for pipes and components;
- Status on oxide dispersion strengthened steel (ODS) as cladding tube material;
- In-service inspection and repair (ISIR) — sensors, inspectability, reparability, robotics.

The main preliminary results are:

- The definition of a large reference core using oxide fuel and characterized by a very low reactivity loss (self-sustainable core) and attractive safety parameters;
- The realization of two irradiation tests in Phénix, concerning structural materials (ODS F/M steel) and oxide fuel elaborated with (U, Pu) co-precipitated powder;
- The definition and the realization of the end-of-life experiments in Phénix in order to accumulate data for the qualification of the codes dedicated to the neutronics and thermohydraulics in the SFR and to improve the knowledge of physical phenomena in an SFR core;
- The characterization of new solutions for the recycling of minor actinides in the heterogeneous mode;
- The first calculation of severe accident sequences with SAS4A and SIMMER multiphysical computational tools;
- Some innovative proposals for new concepts as an IHX/SG integrated component;
- The evaluation of alternative fluids to sodium for the secondary circuit;

- The establishment of the development plan of the technological loops necessary for the R&D and the prototype development;
- The definition and the launch of a comprehensive programme on ISIR;
- Preliminary reactor designs for innovation assessment (pool, loop, fuel handling options, advanced energy conversion systems, etc.);
- Innovative feature proposals aiming to improve the safety cases, in line with the safety authorities' expectations for a Generation IV reactor.

On safety, emphasis is put on the relationship with the safety authority. Interactions started in 2008, including the organization of technical seminars. In 2010, the main topic of exchange will be the feedback on operation of Phénix, Superphénix and other reactors. A special programme is still ongoing to provide accurate analysis of the Phénix situations involving scram due to negative reactivity (AURN) taking into account the end-of-life experiments. Current R&D includes passive safety devices as additional lines of defence in an enhanced core.

As far as the energy conversion system is concerned, the goal is to minimize the frequency and the consequences of sodium–water reaction. R&D is conducted in two directions: (i) an alternative fluid to sodium in the secondary circuit or to steam as a working fluid (Brayton cycle), and (ii) design improvement to improve significantly the resistance to sodium–water reaction, such as modular steam generators with double-wall tubes, etc. Supercritical CO_2 is seen as a long term promising option, with issues such as sodium–CO_2 interaction to be investigated further.

ISIR is of utmost importance and an extensive R&D programme is being performed. The following steps now include a large refurbishment and construction programme for testing facilities to support R&D activities and ASTRID development. Sharing of facilities in other countries is also envisaged.

3. ASTRID PROGRAMME

The ASTRID prototype is seen as an industrial prototype predating the first-of-a-kind, meaning that extrapolation of the technical options and of the safety demonstration is of utmost importance. The reactor will also provide some irradiation capacities, especially in order to validate the expected properties of the new fuel (big pin and ODS cladding) and the capability to burn minor actinides in an industrial manner.

The ASTRID programme defined by the CEA also includes a facility to manufacture the fuel for the reactor, of limited capacity, from 5 to 10 t of heavy metal per year. The refurbishment of existing testing facilities and the construction of new tools is part of the programme as well.

ASTRID will be coupled to the grid with an electrical power of about 600 MW. It will integrate operational feedback on former and current reactors. It is seen as a full Generation IV prototype reactor. Its safety level will be at least as good as current 3rd generation reactors, with strong improvements in core and sodium related issues. After a learning period, the reactor will attain a high load factor (e.g. more than 80%). The reactor will have the capability to demonstrate transmutation of minor actinides on a larger scale than previously done in Phénix. Clearly, the investment costs of the prototype will be kept as low as possible, with technical options compatible for later deployment on a commercial facility.

The schedule associated with the ASTRID prototype is very ambitious and will be adapted in the course of the project, following R&D results and political decisions. First, choices need to be made in 2010 in order to launch the pre-conceptual and conceptual designs, and start preliminary discussions with the safety authority. Still, some options will be kept until 2012. A second phase of conceptual design with the submission of the safety option file in 2014 will allow basic and detailed design to start from 2015. The objective is to put the reactor into operation around 2020.

A PERSPECTIVE ON THE FUTURE DEVELOPMENT OF FBRs IN INDIA

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Abstract

In the Indian energy scenario projections for the future, nuclear power through fast reactors is expected to play an important role, representing ~20% of total installed electrical capacity by 2052. Successful operation of a 40 MW(th)/13 MW(e) capacity fast breeder test reactor over 23 years, strong R&D executed in a multidisciplinary domain and construction of a 500 MW(e) prototype fast breeder reactor (PFBR) based on an indigenous design have provided high confidence in the success of fast breeder technology. Beyond the PFBR, there are plans to construct six more FBRs, each of 500 MW(e) capacity. Towards this end, a systematic roadmap has been drawn up for improved economy and enhanced safety through a number of measures. The major features incorporated to achieve economy are the twin unit concept, plant life increased to 60 years in comparison to 40 years for the PFBR, reduced fuel cycle cost with higher burnup, reduction in the number of steam generators from eight to six, in-vessel primary sodium purification, minimizing the use of 316LN stainless steel for NSSS components, reduction in special steel specific weight requirements, compact plant layout, improved load factor, reduction in construction time by at least 2 years and co-location of the fuel cycle facility. The major features towards enhancing safety features are improvements in the reactor shutdown system to provide reliability for a target of 10^{-7}/reactor-year, enhanced diversity in the decay heat removal system, integrated primary sodium purification, reduction in the number of tubes to tubesheet welds by increasing the seamless tube length of the steam generator, and enhanced in-service inspection. Beyond 2020, a series of 1000 MW(e) capacity metallic fuelled reactors with high breeding potential will be constructed and R&D activities have been systematically planned for metallic fuel development. The paper addresses the highlights of the conceptual design features of future sodium cooled fast reactors in India.

1. INTRODUCTION

India is targeting a per capita electricity consumption of about 2500 kW·h/a by the year 2032 from the current level of about 700 kW·h/a. The Integrated Energy Policy of India forecasts that the installed capacity requirement in 2031–32 would be about 778 GW(e) against present installed capacity of 152 GW(e).

Currently, the share of nuclear power in India is about 3%. However, increasing power demand, maturity in implementing pressurized heavy water reactors (PHWRs), R&D progress towards the fast breeder reactor (FBR) technologies, huge domestic reserves of thorium, the successful conclusion of international civil nuclear cooperation, the attribute of nuclear power in balancing the energy needs and providing energy security are all the opportunity factors promoting the rapid growth of nuclear power in India.

India's three stage nuclear power programme places emphasis on the closed fuel cycle, the first stage of which involves fuel from natural uranium fuelled PHWRs being reprocessed to obtain plutonium. As a logical follow-up in the second stage, plutonium based FBRs are planned so that a sizeable capacity can be established and, eventually, in the third stage, thorium based reactors can be set up and sustained for a long time. Currently known mineable uranium reserves in India can supply only 10 GW(e) capacity based on PHWRs. FBRs employing a closed cycle, in view of its efficient utilization of uranium, is a further option. Further, FBRs are essential for converting thorium to the ^{233}U required for the third stage of the above stated nuclear programme. The closed fuel cycle also permits a significant reduction in the waste for long term management.

2. THE FBR PROGRAMME IN INDIA

The Indian FBR programme started with the establishment of a research centre (then known as the Reactor Research Centre) dedicated to the development of fast reactor science and technology and the decision to construct the fast breeder test reactor (FBTR), in collaboration with France, at Kalpakkam. The FBTR is a sodium cooled loop type 40 MW(th)/13.2 MW(e) experimental reactor and was commissioned in 1985. The experience gained in the construction, commissioning and operation of the FBTR, as well as worldwide FBR operational experience, and 30 years of focused R&D involving extensive testing and validation, material and manufacturing technology development and demonstration, peer reviews and synergism among the Department of Atomic Energy, R&D institutions and industries, have provided the necessary confidence to launch the prototype FBR of 500 MW(e) capacity (PFBR). Reactor construction was started in 2003 and the reactor is scheduled to be commissioned by 2011.

As a follow-up to the PFBR, there are plans to construct six units of 500 MW(e) capacity based on MOX fuel and twin unit design with improved economy and safety. Beyond these seven units, metallic fuelled reactors of 1000 MW(e) unit capacity, with emphasis on breeding, will be deployed.

In the following paragraphs, details of the PFBR, the approach to the design of future FBRs and the roadmap for achieving robust growth of FBR technology with the closed fuel cycle in India are highlighted.

3. THE PFBR AND CO-LOCATED FUEL CYCLE FACILITY

The PFBR has been designed as a techno-economic demonstration of indigenous design and technology and is the forerunner of the series of fast reactors that are planned to be deployed. Co-location of the fuel cycle facility (fabrication, reprocessing and waste management) along with the reactor is also planned so as to minimize the fuel cycle cost of the PFBR and to exercise better control over fuel movement for strategic reasons. The dedicated Fast Reactor Fuel Cycle Facility for reprocessing of PFBR material is designed with additional capacity to handle the reprocessing needs of two more units of 500 MW(e) capacity each.

4. APPROACH FROM THE PFBR TO THE FUTURE FBR

4.1. Economic factors and design approach

Economic competitiveness is vital for commercial deployment of fast reactors. Significant design efforts are necessary to reduce the capital cost of future FBRs coupled with enhanced safety. Therefore, there is a challenge to identify the critical influential parameters that govern the overall cost and safety, and efforts are channelled into optimizing these with focused R&D, keeping in view the international experience. Lessons learnt from the PFBR in terms of plant layout, civil construction, manufacture of NSSS components, in particular technical specifications, tender packages and regulatory review will be incorporated in the design and construction of future FBRs. A detailed review of the capital cost breakdown of PFBR indicates that the reactor assembly, sodium circuits and fuel handling systems require closer scrutiny for possible cost reduction measures and there is little scope in the balance of plant due to the level of standardization and maturity in its associated systems. Apart from the above, the analysis of unit energy costs of the PFBR reveal further tangible benefits for enhancing plant thermal efficiency, fuel burnup and the plant capacity factor, reducing construction time, using multiple unit construction, and introducing policy measures on financial parameters such as depreciation rate, debt equity ratio, interest rate, etc. (Fig.1). Through the above exercise, and also based on

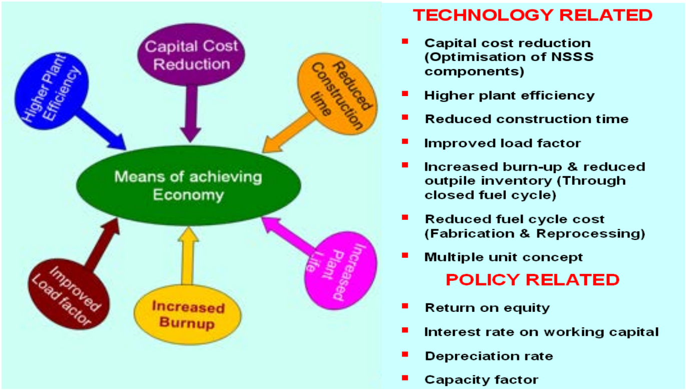

FIG. 1. Factors influencing overall costs.

experience with construction and operation of FBTR and FBR units worldwide and state of the art R&D with respect to sodium cooled fast reactors (SFRs), well-defined optimization objectives/targets are defined for future FBRs, also called the commercial fast breeder reactor (CFBR). These are summarized in Table 1.

There are plans to construct six units each of 500 MW(e) capacity in addition to the PFBR. A systematic roadmap made for the cost reduction of FBRs through the above measures/mechanisms reveals cost benefits of a ~35% reduction in tariff for the CFBR from 3 to 6 in comparison to the PFBR on a constant money basis and with same debt:equity ratio.

4.2. Measures to enhance safety

Improvements in the shutdown system, decay heat removal system, in-service inspection, sodium purification, steam generator and number of primary sodium pipes are some of the key areas identified for enhancing reactor safety.

In the PFBR, two independent and fast acting diverse shutdown systems, comprising nine control and safety rods driven by individual drive mechanisms (CSRDMs) and three diverse safety rods along with their individual drive mechanisms (DSRDMs) are provided. In both systems, the gravity assisted SCRAM action by dropping the absorber rods (CSR/DSR) into the reactor core is achieved through de-energization of the electromagnets holding the rods. Apart from many distinct differences between electromagnets, the location of the electromagnet is also different. In the CSRDM, the electromagnet is located in an argon atmosphere and is housed in the upper part of the mechanism, while in the DSRDM, it is located in the lower part and is immersed in sodium.

TABLE 1. BASIC DESIGN FEATURES

Parameter	PFBR	CFBR
Power (MW(e))	500	500
Coolant (primary and secondary)	Sodium	Sodium
Primary circuit	Pool with external purification	Pool with no primary sodium outrside pool
Fuel	MOX	MOX
Fuel burnup (GW·d/t)	100	150 initially and 200 later
Design life	40 calendar years 75% load factor	60 calendar years 85% load factor
Unit	Single	Twin
Construction time	8 years	8 years
Number of:		
Primary pumps	2	2
Secondary pumps	2	2
IHX/loop	2	2
SG/loop	4	3
SG tube length (m)	23	30
Number of SGDHR loops	4	6
Spent fuel storage medium	Water	Water

Unprotected loss of flow and unprotected transient overpower are two events leading to severe consequences and are initiating events for a core disruptive accident. For the PFBR, each shutdown system consists of a reactor protection system, actuation system and safety support system. Each shutdown system has a non-availability of less than 10^{-3}/reactor-year with the overall objective to achieve a non-availability of both shutdown systems of $<1 \times 10^{-6}$/reactor-year.

For future FBRs, the targeted reliability is 1×10^{-7}/reactor-year for which additional passive/active safety features are considered for implementation in the design. Enhanced reliability of 10^{-7}/reactor-year provides the potential for keeping a whole core accident as a beyond design basis event.

The temperature sensitive magnetic switch and the temperature sensitive electromagnet are two passive safety concepts currently under development. Development work on the temperature sensitive magnetic switch is at an advanced stage. As part of an active system, a stroke limiting device has been

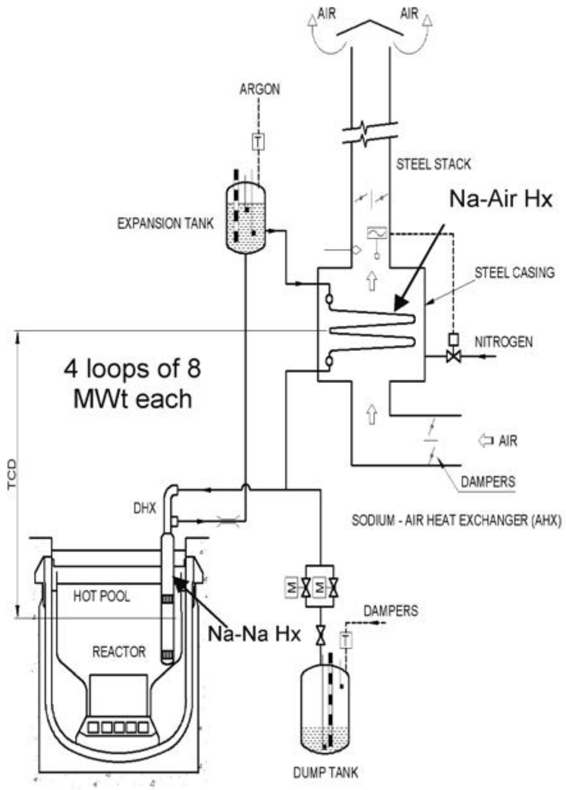

FIG. 2. The safety grade decay heat removal system in the PFBR.

identified for further theoretical and experimental development. Other passive safety features will also be investigated in order to achieve the targeted reliability of shutdown systems.

4.2.1. Decay heat removal system

Accomplishment of smooth decay heat removal, after reactor shutdown, through highly reliable systems is an essential and important safety requirement.

For the PFBR, two different decay heat removal systems are provided, namely, (i) the system operating under normal conditions through the steam–water system, referred to as operation grade decay heat removal system, and (ii) the system whereby decay heat removal occurs by natural convection, intermediate sodium and air paths, and referred to as the safety grade decay heat removal system and designed with adequate diversity. Figure 2 shows the details of the safety grade decay heat removal system in the PFBR.

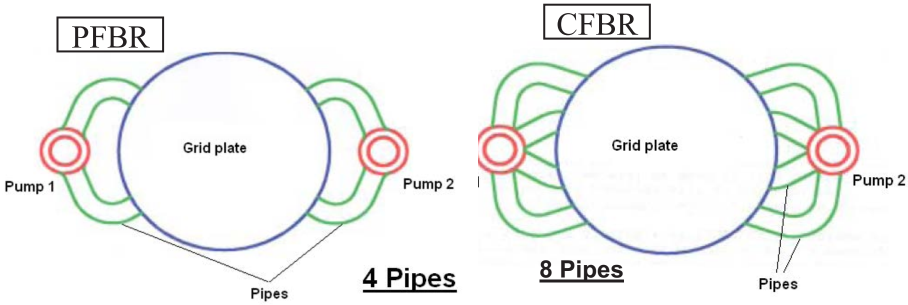

FIG. 3. Primary pipe arrangement.

For future FBRs, it is proposed to provide six dedicated loops with a capacity of 6 MW(th) each. To achieve enhanced diversity in sodium and air flow, it is planned to have three loops working in natural convection mode and the other three loops normally working in forced convection mode with an electromagnetic pump and air blower in each loop, designed to work in natural convection mode with reduced thermal capacity of ~60%.

4.2.2. Arrangement of primary sodium pipes

Four primary pipes feed the primary sodium from the pump to the core in the PFBR (Fig. 3). The pipes are designed as per Class 1 design rules and are made from 316LN stainless steel. Double ended guillotine rupture of one of the primary pipes is considered a Category 4 event as an enveloping case in the PFBR. Plant safety has been demonstrated for this event for the PFBR.

To enhance safety, one of the potential ways is to increase the number of pipes from four to eight (Fig. 3). With the reduction in pipe diameter from 600 mm to 420 mm, the flow in the core increases in the case of a double ended guillotine rupture. Also, owing to the relative decrease in diameter and thickness, the pipe is more flexible in accommodating thermal expansion.

4.2.3. Sodium purification

An ex-vessel sodium purification concept is adopted in the PFBR (Fig. 4), wherein a small quantity of primary sodium is taken out of the main vessel for purification and returned back to the main vessel. The circuit consists of a cold trap, economizer, plugging indicator, priming tank and associated piping. The whole circuit except for the cold trap is kept in a steel cabin filled with nitrogen. For future FBRs, in-vessel purification is being pursued for both safety and economic considerations. This will help in avoiding siphoning of the primary

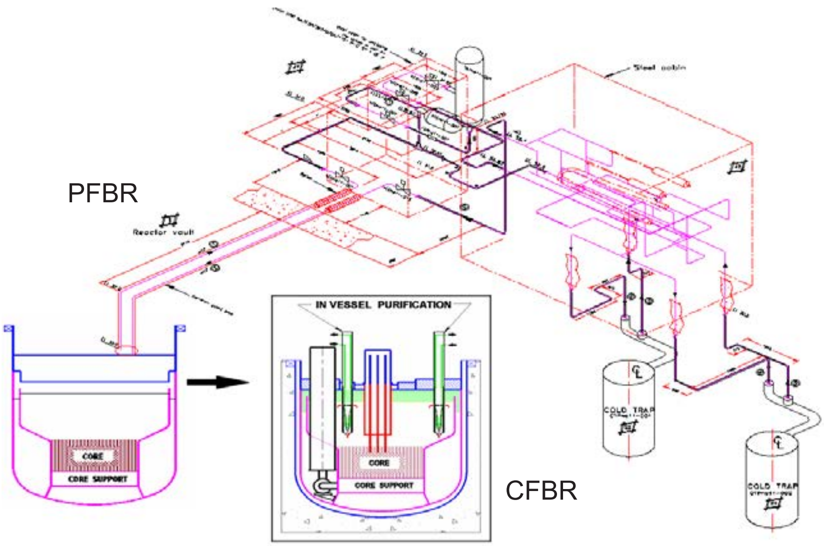

FIG. 4. Primary sodium purification.

sodium and minimize the probability of primary sodium leaks. Design and development of a cold trap suitable for in-vessel purification is under progress.

4.3. NSSS and component design

A close review of the design aspects of major components is in progress. Significant conceptual changes have been made in a few of the major components based on the design and manufacturing experience with the PFBR. For example, the changes in the arrangement of shielding around the core, reactor assembly, sodium circuits, fuel handling, construction materials, in-service inspection, etc., are highlighted briefly below.

4.3.1. Radial shielding around the core

Shields around the core and blankets form a major part of the reactor assembly in fast reactors. Boron carbide and stainless steel have been the main choice shield materials for shields in fast reactors. The PFBR core is configured with 609 stainless steel (in six rows) and the radial shielding with 417 boron carbide (in three rows). For the CFBR, it is proposed to use ferroboron as a shield material for economic considerations. Commercially available ferroboron in the form of lumps, granules and powder has 15–18 wt% boron and a bulk density of around 4 g/cm^3. Detailed calculations indicate that eight rows of shielding

TABLE 2. SUMMARY OF OPTIONS FOR THE CFBR

Component/feature	PFBR	Options being studied for CFBR
Top shield	Welded box structure filled with high density concrete	Welded box structure filled with higher density concrete — reduced height Thick plate — reduced height Formed dished head with external shielding
Complementary shielding over top shield	• Formed shells • Larger gap due to form tolerance • Large complementary shielding	• Machined penetration/stepped construction • Smaller gap • Reduced complementary shielding
Grid plate	Large diameter grid plate (botted)	• Smaller grid plate (welded) • Shell enveloping core subassemblies that are cooled • Support for peripheral shielding subassemblies that are not cooled through spikes
Support for reactor assembly	• Through an extended shell (~1.5 m) in tension • Increased component height over top shield	• Through shell in compression • Reduced component height over top shield

assemblies with ferroboron meet the radial shield requirements satisfying the radiological safety criteria. The effectiveness of ferroboron stems from the fact that boron is spread throughout the shield region, though in lower atomic densities, and iron present in the shield regions also contributes significantly to attenuation. Measurements of thermal, epithermal and fast neutron attenuation in ferroboron have been carried out in the KAMINI reactor for comparison with boron carbide, which have indicated the effectiveness of ferroboron as a shield material. As part of the development work on ferroboron, diffusion experiments under accelerated test conditions are in progress.

4.3.2. Reactor assembly

Significant design changes are being contemplated in the design of major reactor assembly components with a view to optimizing the design and reducing capital cost. Further, the manufacturing experience with components for the PFBR has highlighted the focus areas that need simplification. Table 2 broadly

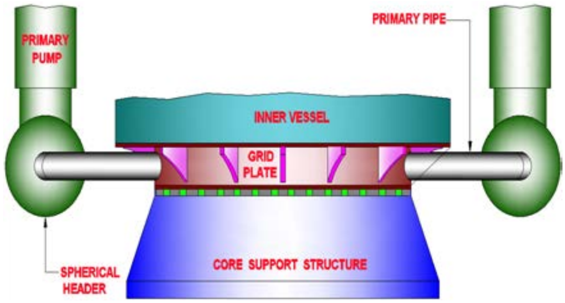

FIG. 5. Grid plate arrangement in the CFBR.

summarizes the options being studied for the CFBR. For example, the adoption of a fully welded design concept for the grid plate (Fig. 5) with a reduced number of sleeves that support only the core subassemblies requiring coolant flow and a spike supporting arrangement for other peripheral subassemblies has offered considerable size and economic advantages, reflected in overall manufacturing time as well. These changes have resulted in a 55% reduction in the overall weight of the grid plate. For reducing the shielding in annular gaps between rotatable plugs and the roof slab in the top shield, thick plates with machined gaps offer a potential solution. The other changes that are being actively pursued include options for a roof slab, change of configuration in the support of the reactor assembly, etc.

4.3.3. Heat transport and auxiliary systems

Optimization of piping layout with a view to minimizing the overall length and number of welds, closer assessment of margins in system capacities, review of the number of valves and piping supports in the sodium and auxiliary systems and optimization of steam generator design for a reduced number of tube to tube sheet welds, etc., are some of the measures being thoroughly reviewed for the CFBR. The PFBR steam generator was designed with a tube length of 23 m. However, for the CFBR, the longer length of 30 m is preferred in order to reduce the number of tube to tube sheet joints by ~35%, thus minimizing the possibility of sodium–water reaction throughout the design life, reducing manufacturing time and improving economics (Fig. 6). Detailed studies, taking into consideration the overall effect of capital cost, outage cost and construction schedule, have indicated that a design with 3 modules per loop each is optimum.

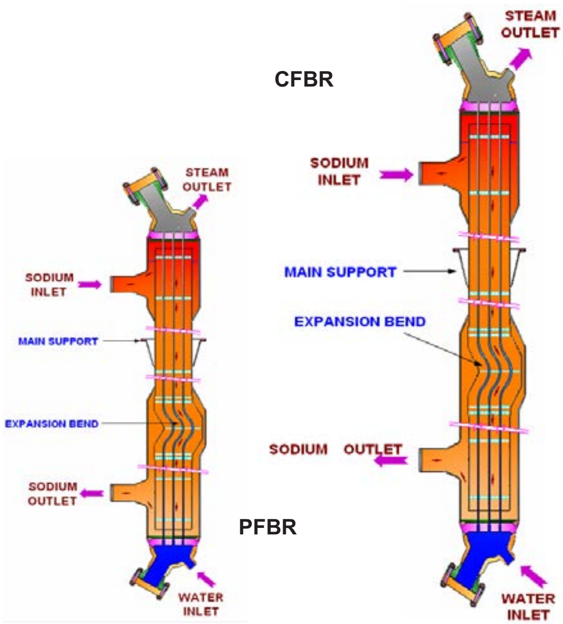

FIG. 6. Steam generator.

4.3.4. Fuel handling system

The fuel handling system in the PFBR uses a combination of two rotatable plugs, an offset arm type in-vessel handling machine (transfer arm), an inclined fuel transfer machine for ex-vessel handling and a water pool for ex-vessel storage of subassemblies. For the CFBR (Fig. 7), the single ex-vessel water pool storage is retained with the number of storage locations optimized to meet the normal storage requirements of both units and emergency full core unloading of one unit. Further simplification of the in-vessel and ex-vessel handling schemes is also being attempted by the use of a flask type transfer in place of the inclined fuel transfer machine; the handling flask being common to both the units. The offset arm type concept is retained for in-vessel handling and two such machines will be utilized.

The sodium cleaning and decontamination systems are housed inside the reactor containment building in the PFBR. For the CFBR, significant reduction in the cost of the fuel handling system is envisaged by sharing the fuel handling and decontamination facilities between both the units. The decontamination system is also located outside the reactor containment building in a separate building common to both the units. The use of water inside the reactor containment building is thus minimized, resulting in enhanced safety.

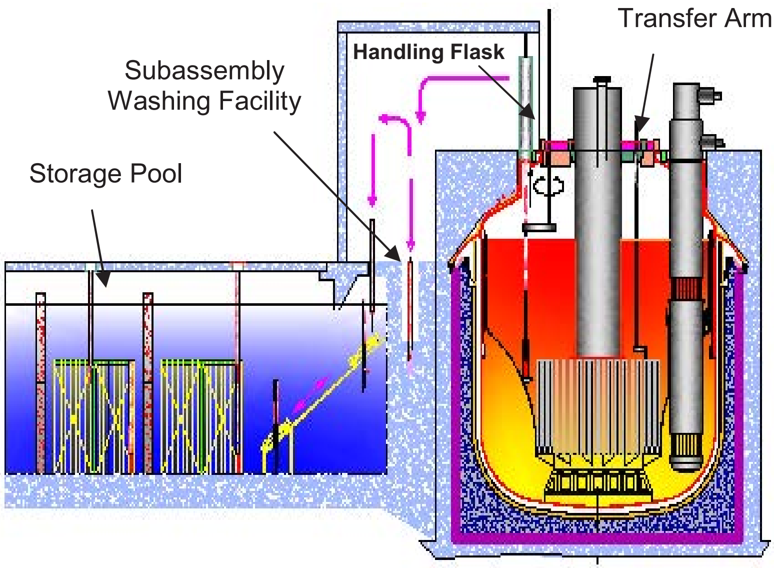

FIG. 7. The CFBR fuel handling system.

4.3.5. Other design measures

The options considered for reducing the size and cost of the reactor assembly are:

(a) Three thermocouples at each subassembly outlet against two in the PFBR;
(b) Study of a 'bean shaped' intermediate heat exchanger (IHX) in order to reduce the radial width of the roof slab required to accommodate the IHX support flange (Fig. 8);
(c) Capability to handle short cooled fuel (with higher decay heats) for faster recycling of spent fuel.

4.4. Construction materials

With a view to enhancing the fuel burnup progressively, use of 9Cr–1Mo for the wrapper, improved austenitic stainless steel 20% cold work D91 for cladding in the first few cores for burnup of up to 150 GW·d/t and subsequently oxide dispersion strengthened steels for burnup of up to 200 GW·d/t is envisaged. The use of 304LN in place of 316LN for cold leg near core components after material confirmation studies for higher plant life is the other area of focus for reducing material costs of future FBRs (Table 3).

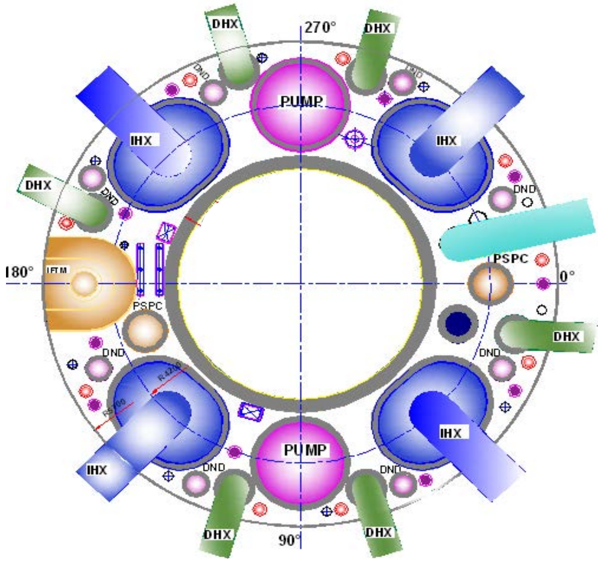

FIG. 8. Top shield layout with bean shaped IHX.

4.5. In-service inspection

In-service inspection plays a major role in monitoring the condition of nuclear power plant components and allows appropriate corrective measures to be taken if needed. Owing to the presence of high radiation levels, automation plays a vital role in the design of in-service inspection equipment. For the inspection of the PFBR's main vessel and safety vessel (nominal gap of 300 mm), an automated free roving four wheeled device is provided with facilities for on-board systems for visual examination/navigation purposes and an ultrasonic examination device for volumetric examination. Similarly, for the critical examination of steam generator tubes, a four legged walking robot has been indigenously developed.

For future FBRs, an in-service inspection device that will work within a smaller 200 mm nominal gap between the main vessel and safety vessel is planned with gas coupled ultrasonic testing using electromagnetic acoustic transducers or phased array ultrasonic testing using microelectromechanical systems. Development of examination and crack detection/repair under sodium is also planned.

TABLE 3. CONSTRUCTION MATERIALS

Component	PFBR	CFBR
Clad	20% CW 15Cr-15Ni + Mo + Ti + Si ASTMA 771	20% CW 15Cr-15Ni + Mo + Ti + Si + B + P
Wrapper	-do-	9Cr–1Mo
Main vessel	316LN	316LN
Safety vessel	304LN	Carbon steel (A48P2)
Grid plate Core support structure	316LN	Study for 304LN
IHX	316LN	316LN
Steam generator	Modified 9Cr–1Mo (Gr.91)	Gr.91
Secondary sodium piping	316LN	Study for Cr–Mo Linked to availability of sodium valves in Cr–Mo
Sodium pumps, sodium tanks	304LN	304LN

5. PLANT LAYOUT FOR TWIN UNITS

Figure 9 shows the proposed layout for the twin units. Emphasis is placed on achieving greater compactness and the sharing of buildings/facilities wherever possible without compromising safety. The fuel building, radwaste building, decontamination building, control building, electrical building, site assembly shop and the non-safety related plant services, other than the turbine building, are shared. For future FBRs, the decontamination system is shifted outside the reactor containment building and is located as a common facility for the twin units enhancing safety by minimizing the use of water inside the reactor containment building. The reactor containment buildings, steam generator buildings and fuel buildings will be put on a common base raft. Layout of the buildings for the reactor will be planned as symmetrical.

6. BEYOND THE PFBR: THE ROADMAP FOR THE WAY FORWARD

Considering the current status, industrial infrastructure and R&D needs of the future programme and keeping to the timescale for planned growth of nuclear energy, a strategy for the deployment of FBRs over the short, medium and long terms has been evolved.

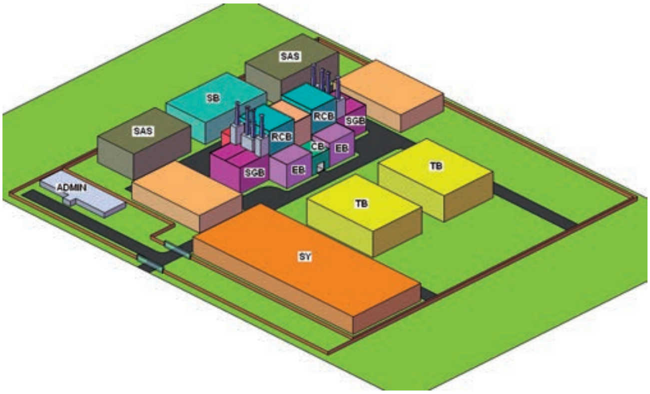

FIG. 9. Plant layout for twin units.

In the short term, the deployment of six more MOX based FBRs is planned in order to achieve the commercial maturity of FBRs in the country (Table 4). The reactor units will be co-located with fuel cycle facilities both for economic and security reasons. The fuel for the reactors will be sourced from indigenous PHWR spent fuel.

Besides the six MOX based FBRs, metallic fuel based FBRs (MFBR) will also be launched. The first MFBR of 1000 MW(e) capacity is expected to be in operation by 2027. Evolution of the FBR installed capacity is linked to a number of factors, including the indigenous thermal reactor programme, thermal reactors under international cooperation, metal fuel design, cooling period of spent fuel prior to reprocessing and reprocessing loss.

TABLE 4. SCHEDULE OF MOX BASED FBRs

Project	Unit capacity (MW)	Project capacity (MW)	Year of commercial operation	Location
PFBR	500	500	2012	Kalpakkam (co-located with the fast reactor fuel cycle facility)
FBR 1 and 2	500	1000	2020	
FBR 3 and 4	500	1000	2022	To be finalized
FBR 5 and 6	500	1000	2023	

TABLE 5. METALLIC FUEL IN THE FBR

Properties	Sodium bonded			Mechanical bonded U–Pu–zircaloy liner
	U–Pu–10%Zr	U–Pu–6%Zr	U–Pu plus zircaloy liner	
Experience	EBR-II + FFTF	Limited EBR-II		Limited with Cr/W coating in BOR-60 and BN-350
Breeding ratio	1.36	1.47	1.56	1.56
Reactor doubling time (a)	10	7.6	6.6	6.6

6.1. Metallic fuel reactors

The most important aspect of future nuclear power growth will be through the use of metallic fuel FBRs, with emphasis on breeding and thus shorter doubling time. There are four types of design configuration considered for development: three sodium bonded and the fourth mechanically bonded.

With sodium bonded metallic fuel, the three designs under consideration are: (i) U–Pu–10%Zr, (ii) U–Pu–6%Zr and (iii) U–Pu plus zircaloy as liner (Table 5).

The roadmap for metallic fuel development is through testing of pins, testing of the 37 pin subassembly in the FBTR, conversion of the FBTR to a metallic fuel core and testing of a few subassemblies in the PFBR pyrochemical reprocessing of metallic fuel to take the best advantage of short cooled fuel and this has been recognized as an important R&D programme. There is also a plan to construct a medium sized (~300 MW(th)) metal fuel test reactor for carrying out irradiation studies on fuel subassemblies of power reactor size. Also, one of the six 500 MW(e) MOX fuelled FBRs will be designed to have a flexible core to accept both MOX and metal fuels.

7. SUMMARY

The FBR employing a closed fuel cycle is the obvious technology option to pursue towards providing energy security for India. The PFBR is a technocommercial reactor and economic competitiveness is important for rapid commercial deployment of FBRs subsequent to the PFBR. Towards this end, design studies intended to achieve enhanced safety and economy for future FBRs, with a targeted plant life of 60 years and construction time of 60 months, are being implemented. Enhanced safety is proposed to be achieved through several design provisions that include adopting additional passive and active safety features in the shutdown system, improving the reliability of the decay heat removal system by enhancing to six loops, improving in-service inspection, adopting in-vessel sodium purification, increasing the number of primary pipes to eight, increasing the length of steam generator tubes and thereby reducing the number of tube to tube sheet welds, minimizing the use of water inside the reactor containment building by shifting decontamination facilities outside, etc.

Economy improvement will be achieved by increasing burnup, increasing plant life to 60 years, increasing the plant load factor to 85%, reducing the cost of reactor assembly, optimizing design solutions for sodium circuits and fuel handling systems, optimizing plant layout through sharing of facilities and by reducing the construction time. Further, the roadmap for large scale deployment of the FBR towards ensuring domestic energy security is detailed through the use of metallic fuel reactors with emphasis on breeding gain and co-located fuel cycle facilities based on pyrochemical reprocessing.

RESEARCH AND DEVELOPMENT POLICY ON FBR CYCLE TECHNOLOGY IN JAPAN

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Abstract

The fast breeder reactor (FBR) is a quite effective and realistic measure for establishing a long term, stable energy supply and for preventing global warming. In Japan, the FBR research and development project, named FBR Cycle Technology Development (FaCT), has been operational since April 2006. In this project, the combination of a sodium cooled fast reactor using oxide fuel and advanced aqueous reprocessing, as well as the simplified pelletizing fuel fabrication, is being developed principally as the most promising concept of FBR cycle technology to be commercialized, aiming at introducing the demonstration FBR by around 2025, and the commercial FBR before approximately 2050. Research and development for the establishment of the innovative technologies, which can meet design requirements for the demonstration FBR, has been steadily progressing. The adoption of the innovative technologies will be decided by judging their applicability and the conceptual designs of demonstration and commercial FBR cycle facilities by 2015. Consequently, the development of innovative technologies should be completed by 2015. Thereafter, the FaCT project will enter the introduction stage through a system demonstration.

1. INTRODUCTION

Today, humankind faces global issues on a scale never seen before, including global warming and energy resource security. Under such circumstances, nuclear energy is important for solving the energy problem and global climate change simultaneously, through the realization of a stable energy supply and zero CO_2 emissions. This idea is gaining recognition all over the world and, accordingly, Japan has promoted research, development and use of nuclear energy as the major source of electrical power. Further, it is aimed at the establishment of the FBR and its fuel cycle (hereafter referred to as the FBR cycle), which will ensure a long term energy supply through the efficient use of uranium resources.

The technologies for the FBR cycle can not only achieve dramatically efficient utilization of uranium resources but also burn minor actinides recovered through reprocessing of spent nuclear fuels. The latter can reduce the amount of

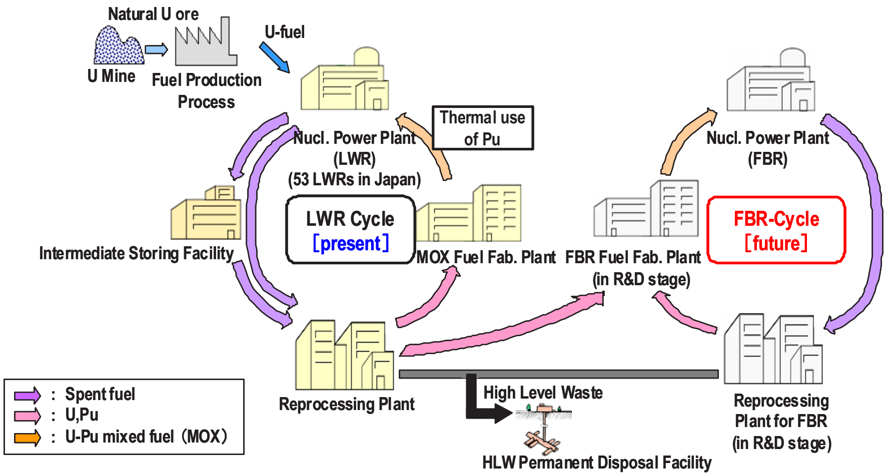

FIG. 1. Status of nuclear fuel cycle in Japan.

high level radioactive waste and improve the proliferation resistance. If such FBR cycle technology can be available in a safe and economical manner similar to that of light water reactor (LWR) technology, it may contribute to sustainable human development as well as a stable energy supply.

Currently, the thermal use of plutonium is in progress in Japan (Fig. 1). The construction of the reprocessing plant will be completed by October 2010. As for the radioactive waste disposal issue, the selection of the candidate site for the permanent disposal of high level waste is in progress at the Nuclear Waste Management Organization of Japan.

For commercial FBRs, it will be aimed for introduction before around 2050 (Fig. 2) on the premise of meeting the necessary conditions, such as its economic viability. At this point, it is necessary to consider the progress of nuclear fuel cycle projects for LWRs, the study for commercialization of FBRs, reflecting the experience of the operation of the prototype FBR Monju and the supply and demand situation for uranium.

2. JAPANESE BASIC POLICY ON FBR CYCLE DEVELOPMENT

In Japan, an important issue is to ensure a long term, stable energy supply by establishing FBR cycle technology. FBR cycle technology is considered one of the key technologies of national importance in the third term Science and Technology Basic Plan (JFY2006–2010) [1]. This means that FBR cycle

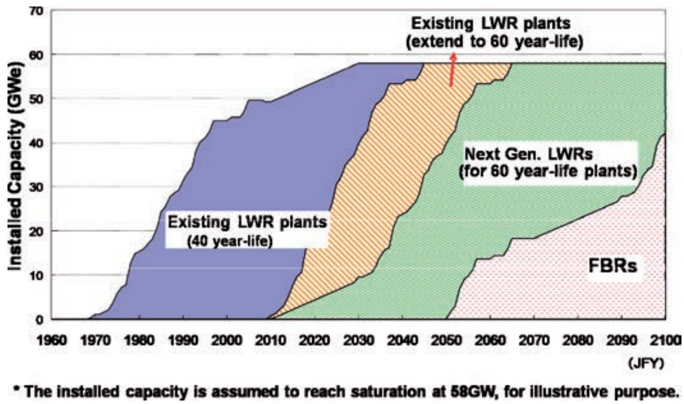

FIG. 2. Long term perspective of nuclear energy in Japan.

technology is recognized as an essential technology for investing intensively in a large scale national project during the period of the basic plan.

In the Framework of Nuclear Energy Policy [2], which is the foundation of Japanese policy on research, development and utilization of nuclear energy, it is stated that research and development of FBR cycle technology should be promoted towards achieving commercialization. This would make it is possible to realize a long term, stable energy supply and a reduction in latent radiotoxicity of radioactive waste by implementing appropriate research and development plans to commercialize the FBR cycle by around 2015 and to introduce a commercial FBR by around 2050.

In Japan's Nuclear Energy National Plan [3], it is stated that the research and development of the FBR cycle technology should be promoted with best efforts for its early commercialization. It also states that the demonstration FBR is aimed to start operation by around 2025 and that the necessary demonstration processes will be performed using the demonstration FBR, that the commercial FBR cycle system will be introduced before around 2050 and that, thereafter, existing LWRs in the end of life will be replaced by FBRs one by one.

3. FRAMEWORK OF FBR CYCLE DEVELOPMENT

On the basis of the Japanese basic policy for the development of FBR cycle technology, the Japan Atomic Energy Agency (JAEA) launched the FaCT project in cooperation with the Government, the electrical power companies and the electrical manufacturers. Figure 3 shows the outline of the development plan leading to the commercialization of the FBR cycle technology in Japan. In the FaCT

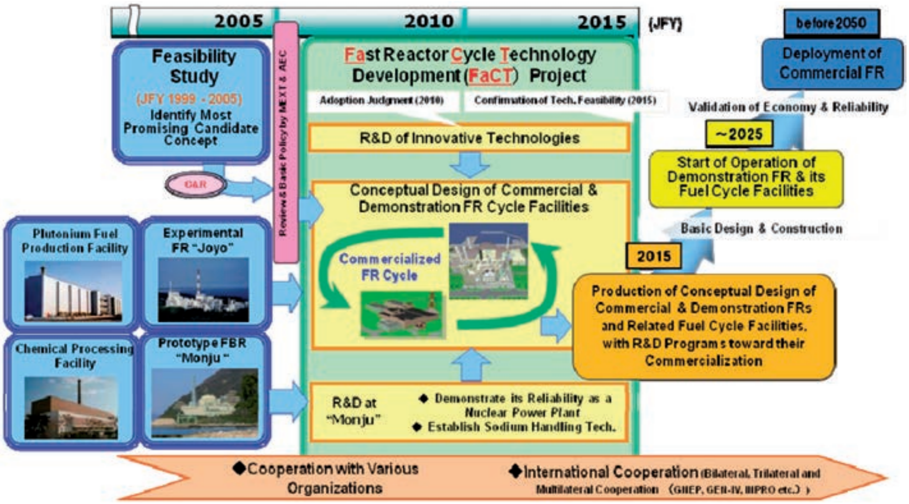

FIG. 3. Outline of development plan for commercialization of FBR cycle technology in Japan.

project, a design study and the research and development of innovative technologies are in progress to determine the adoption of innovative technologies in 2010 and to present the conceptual designs of demonstration and commercial facilities by 2015. Thereafter, the FaCT project will enter the introduction stage through a system demonstration. It is planned to introduce the demonstration FBR by around 2025 and the commercial FBR before around 2050.

In the FaCT project, the experimental fast reactors Joyo and Monju play important roles. After the restart, expected by the end of March 2010, Monju will achieve the initial goals set in the first decade, which are reliable demonstration as a power plant and establishment of the sodium handling technology. The results of Monju are to be reflected in the design of the demonstration FBR. Joyo will also be used for the irradiation of fuels and materials for the improvement of safety and economic competitiveness of the FBR.

3.1. The FaCT project

Prior to the FaCT project, the feasibility study was carried out from 1999 to 2006 by a Japanese joint project team comprising the JAEA and the Japan Atomic Power Company, aiming to present both an appropriate route to commercialization of the FBR cycle technology and its research and development programme by 2015 [4, 5]. In this study, conceptual design features were evaluated to select the FBR cycle systems that could meet the design requirements,

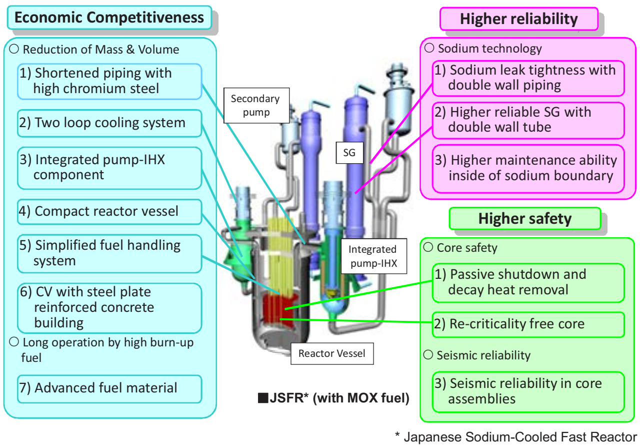

FIG. 4. Innovative technologies of the sodium cooled FBR system.

which were embodied in the five development targets: (i) safety, (ii) economic competitiveness, (iii) efficient utilization of nuclear fuel resources, (iv) reduction of environmental burden, and (v) enhancement of nuclear non-proliferation. As a result, the combination of a sodium cooled FBR using oxide fuel, advanced aqueous reprocessing and simplified pelletizing fuel fabrication was selected as the most promising concept of the FBR cycle system [6–11].

On the basis of the results of the feasibility study, production of the conceptual design for commercial and demonstration FBRs and related fuel cycle facilities with the research and development programme for their commercialization are to be carried out in the FaCT project.

Although the sodium cooled FBR selected as the most promising concept of the FBR system by the feasibility study has already benefited from the fundamental knowledge gained with respect to the technologies, design, construction and operation of the Joyo and Monju plants. However, it is necessary to develop the advanced loop concept, adopting innovative technologies such as the two loop cooling system and the integrated pump–intermediate heat exchanger component and so on, in order to reduce the amount of plant materials and to achieve high levels of economic competitiveness. The innovative technologies in the research and development on the design of the sodium cooled FBR system are shown in Fig. 4.

Most of the development issues for the innovative technologies are aiming at high economic competitiveness by reducing construction and fuel cycle costs and improving plant availability. Research and development on these technologies has been steadily promoted. The fundamental issue, which is very difficult or takes a long time to develop and needs to be substituted with the existing technologies, is not recognized.

3.2. Monju status and restart schedule

Monju, which was designed on the basis of the findings from Joyo, achieved initial criticality in 1994. It has a role in confirming the technological base for the design and safety and in accumulating the operational experiences of sodium cooled FBRs leading to commercialization. The operation, however, has been suspended since a sodium leak incident occurred at the secondary cooling circuit in 1995. After comprehensive reviews of the validity of FBR development in Japan, as well as the safety of Monju, the plant modification work to guard against sodium leakage was carried out between 2005 and 2007. Seismic safety for Monju was evaluated on the basis of the experience gained from the Chuetsu-oki earthquake at Niigata in July 2007; the standard seismic strength for Monju was modified from 466 gal to 760 gal, and a new seismic strength is now being confirmed for maintaining seismic safety for facilities and equipment. The administration structure was reinforced to overcome the weakness in management, revealed in the troubles with sodium leak detectors and outdoor ventilation ducts. The regulatory authority approved the significant advancement in July 2009. The soundness of the facilities and equipment was also evaluated, the outdoor ventilation duct was repaired in May 2009, and the entire system function test was completed in August. System startup test preparation and inspection will be completed in January 2010. On the basis of these achievements, restart is expected by the end of March 2010.

After the restart of Monju, the initial goals of demonstrating its reliability as an operational power plant and establishing sodium handling technology will be given priority. Then, Monju will be used to further develop high performance FBRs. It is considered that Monju should be used as a base for international research and development in the field of FBR cycle technology by making the plant available for joint use for conducting performance tests and by contributing facilities. The validity of design methods for FBR plants can be confirmed, and their accuracy can be improved by using actual operational data from Monju. Improving design margin can contribute to the enhancement of the safety and economic competitiveness of demonstration FBRs and commercialized FBRs. Monju is expected to be used as a site for the human development of future FBRs through operational and maintenance expriences.

4. INTERNATIONAL COOPERATION

International cooperation plays an important role in the development of FBR cycle technology, as its development actually requires a long term effort and a large number of resources. Japan has been promoting multilateral/bilateral international cooperation, such as the Generation IV International Forum (GIF), the JAEA–CEA–USDOE[1] trilateral collaboration on sodium cooled fast reactor (SFR) demonstration/prototypes, and the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), etc., in order to establish fast reactor cycle technologies using global standard technologies and to develop them efficiently.

The GIF is a representative, multilateral collaboration framework. Japan has participated in this framework since the initial stage of GIF and is a chair country of the policy group. Japan has actively cooperated on SFR systems as a chair country of the system steering committee.

The JAEA, CEA and USDOE have a shared vision for research and development aimed at the commercialization of SFRs and have contributed to the development of successful demonstration/prototype SFRs.

INPRO is a multilateral collaboration framework of the IAEA. Japan has participated in a joint study to assess an innovative nuclear energy system based on a closed nuclear fuel cycle with fast reactors and has implemented the assessment study on the Japanese FBR cycle concept by using the INPRO assessment method.

On the basis of the above mentioned arrangements, Monju has accepted a number of researchers from the CEA, EDF and the USDOE. They analysed the existing Monju operational data on sodium thermohydraulics [12, 13], core physics [14], fuel handling and in-service inspections.

It is expected that research and development can be more efficient, reducing the risk and burden on resources by sharing basic data and infrastructure through active multilateral/bilateral international cooperation. Japan will promote research and development for the FBR cycle through the utilization of international cooperation.

5. CONCLUSIONS

The development of the FBR cycle technology in Japan has been promoted as the main concept, which is as follows: combining a sodium cooled FBR using

[1] CEA: Commissariat à l'énergie atomique; USDOE: United States Department of Energy.

oxide fuel and advanced aqueous reprocessing as well as the simplified pelletizing fuel fabrication, adopted with innovative technology. The development will be advanced to a new stage after the restart of Monju. The use of operational data for Monju and operational experience and human development is expected to contribute to the enhancement of the safety and economic competitiveness of the demonstration FBR.

Under the FaCT project now in progress, Japan will promote the design study, and research and development of innovative technologies aimed at deciding on the adoption of innovative technologies by evaluating their applicability in 2010 and presenting conceptual designs for commercial and demonstration facilities by 2015, with a view to starting operations by around 2025 and introducing the commercial FBR cycle system before around 2050.

Furthermore, international collaboration plays an important role in the development of the FBR cycle technology, as its development requires long term efforts and a large number of resources. It is expected that the Monju site will become an international base for research and development in the field of FBR cycle technology.

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STATUS OF FAST REACTOR AND PYROPROCESS TECHNOLOGY DEVELOPMENT IN THE REPUBLIC OF KOREA

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Abstract

A fast reactor system with pyroprocess technology is one of the most promising options for electricity generation, with an efficient utilization of uranium resources and a reduction of radioactive wastes. On the experience gained during the development of the conceptual designs for KALIMER reactors, Korea Atomic Energy Research Institute (KAERI) is currently developing advanced sodium cooled fast reactor (SFR) design concepts that can better meet the Generation IV technology goals. The long term SFR development plan will be carried out with the aim of constructing an advanced SFR demonstration plant by 2028. For the development of pyroprocess technology, KAERI is currently establishing a **pyr**oprocess **i**ntegrated inactive **de**monstration facility (PRIDE), a mock-up facility for pyroprocessing, to produce the engineering data to be incorporated into the design of an engineering scale pyrochemical process facility, which is scheduled to be constructed by 2016.

1. THE DOMESTIC NUCLEAR POWER PROGRAMME

The role of nuclear power in electricity generation is expected to become more important in the years to come in achieving energy self-reliance for the Republic of Korea because nuclear energy is less dependent on natural resources. As of December 2008, there are now 16 PWRs and 4 PHWRs in operation. According to The Fourth Basic Plan for Long-term Electricity Supply and Demand, four OPR-1000s (two at Kori and two at Wolsong) and two APR-1400s at Kori are currently under construction, and six additional APRs will be constructed by 2022.

For the time being, PWRs will remain the major source of nuclear power in the Republic of Korea. However, the storage of the spent fuels produced by these PWRs is a big issue. The on-site spent fuel storage capacity will reach its limit by 2016. Therefore, a decision making process for spent fuel management is under way.

For the safe management of radioactive wastes, including spent fuel, the National Assembly passed the Radioactive Waste Management law on

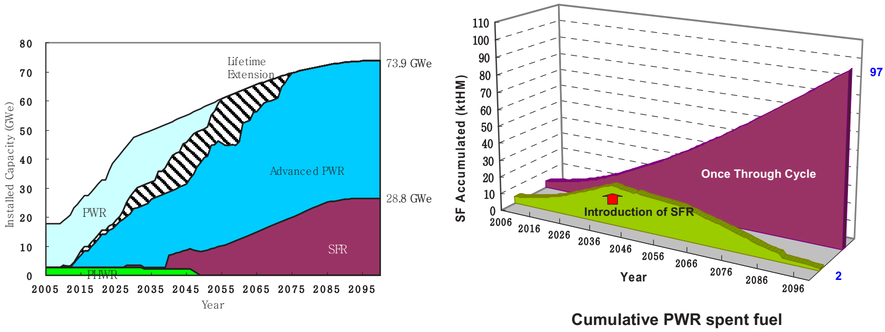


FIG. 1. Reactor transition scenario and spent fuel inventories.

26 February 2008. According to this law, a basic plan for radioactive waste management, with the approval of the Korea Atomic Energy Commission (KAEC) is on going. The Korea Radioactive Waste Management Corporation was established in January 2009 and radioactive waste management funds were established under its administration. In order to examine the spent fuel management plan under public consensus, Korean Public and Stakeholder Engagement was proclaimed in September 2008 by the Ministry of Knowledge Economy.

A favourable neutron balance feature in a fast reactor design makes flexible waste management strategies possible by introducing fast reactors with an appropriate conversion ratio. Fuel cycle impacts resulting from the introduction of sodium cooled fast reactors (SFRs) in the existing PWR dominant nuclear fleet were evaluated by the Korea Atomic Energy Research Institute (KAERI) to establish an efficient reactor deployment strategy applying to a newly increased nuclear power share in the national energy plan. In this way, the PWR spent fuel disposal is reduced by 98% as shown in Fig. 1 and the cumulative uranium demand for PWRs until 2100 is projected to be 821 ktU with a 96 ktU saving. The SFR mix ratio in the nuclear fleet near the year 2100 is estimated to be approximately 39%.

2. SFR AND PYROPROCESS TECHNOLOGY DEVELOPMENT PROGRAMME

The commencement of domestic SFR technology development efforts dates back to 1992 and basic research was performed before 1997. The conceptual design of KALIMER-150 and basic technologies had been developed between 1997 and 2001. The conceptual design of the KALIMER-600 [1] was developed

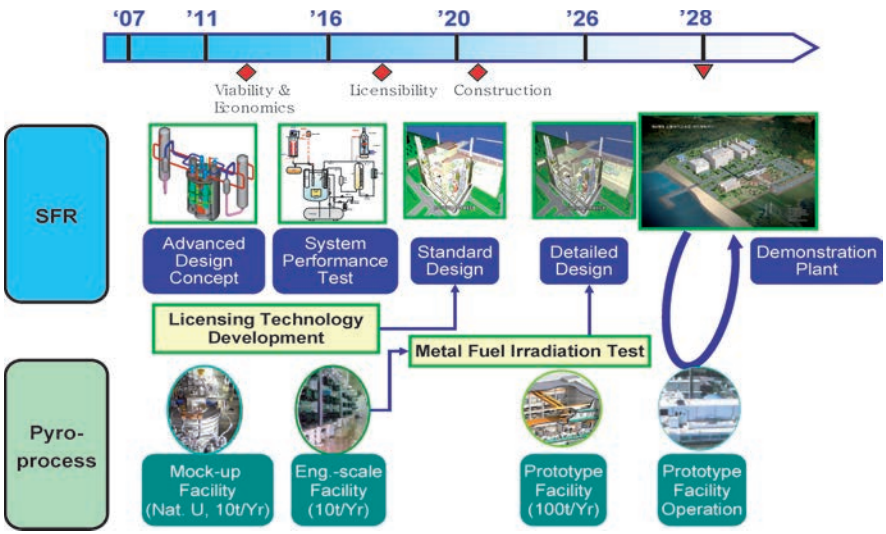


FIG. 2. Long term plan for SFR and pyroprocess technology development.

between 2002 and 2006. KAERI is currently developing the advanced SFR design concepts that can better meet the Generation IV technology goals.

In order to provide a consistent direction to long term R&D activities, KAEC approved a long term development plan on 22 December 2008 for future nuclear reactor systems which include SFRs, pyroprocess technology and the very high temperature reactor. This long term plan will be implemented through the nuclear R&D programmes of the National Research Foundation, with funds from the Ministry of Education Science and Technology. A detailed implementation plan is now being developed.

The long term SFR development plan approved by the KAEC will be carried out with the long term vision of constructing an advanced SFR demonstration plant by 2028 in association with pyroprocess technology development in three phases as shown in Fig. 2:

(1) First phase (2007–2011): Development of an advanced SFR design concept;
(2) Second phase (2012–2017): Standard design of an advanced SFR plant;
(3) Third phase (2018–2028): Construction of an advanced SFR demonstration plant.

For the development of pyroprocess technology, KAERI is currently establishing a **pyr**oprocess **i**ntegrated inactive **de**monstration facility (PRIDE), a

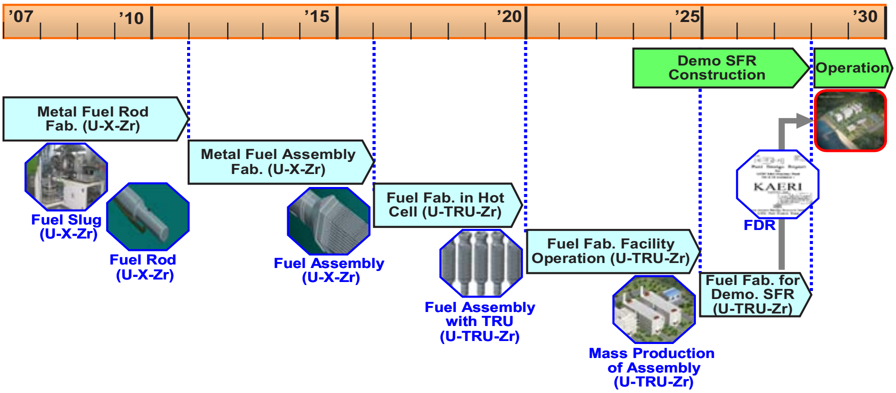


FIG. 3. Long term plan for metal fuel.

mock-up facility for pyroprocessing, to produce the engineering data to be incorporated into the design of an engineering scale pyrochemical process facility, which is scheduled to be constructed by 2016. The Korea Advanced Pyroprocess Facility, a pilot scale facility, will be constructed by 2025.

The pyroprocessing technology capitalizes on the recovery of actinide elements from spent fuel for recycling and fissioning in SFRs for the purpose of power generation. The overriding goal of this R&D plan for pyroprocessing technology combined with SFRs is to develop a closed nuclear fuel cycle that is economically viable, resistant to diversion of nuclear materials for a nuclear weapons programme, and that minimizes the generation of waste products, thereby efficiently increasing the capacity of a final spent fuel repository approximately one hundred-fold. In this fuel cycle, plutonium remains with other isotopes and impurities throughout the processes and cannot be chemically separated in a pure form, which reduces the risk of nuclear proliferation. Confining the final product in a hot cell also makes it far less open to misuse.

The long term plan for metal fuel will be also carried out according to the long term plan for SFR and pyroprocess development, as shown in Fig. 3. Metallic fuel development for SFRs started in 2007 in order to develop high burnup metal fuel. The U-X-Zr metal fuel rod and the U-X-Zr metal fuel assembly will be fabricated by 2011 and 2016, respectively. The U-TRU-Zr[1] metal fuel assembly in combination with pyroprocess facilities will be fabricated remotely in 2020. The mass production of the U-TRU-Zr metal fuel assembly will be followed for the operation of the demonstration SFR from 2025.

[1] TRU: transuranic elements.

3. SFR TECHNOLOGY DEVELOPMENT

3.1. Advanced concept design studies

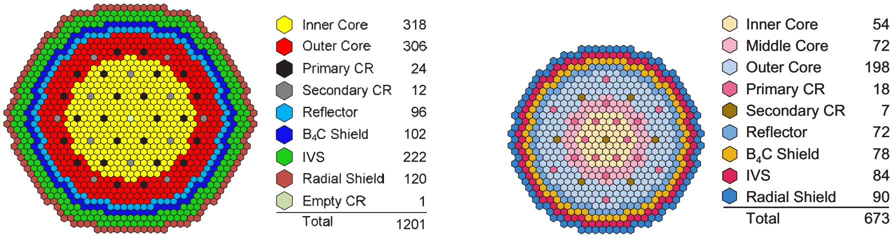
Various advanced design concepts have been proposed and evaluated against the design requirements which were established to satisfy the Generation IV technology goals of sustainability, safety and reliability, economics, proliferation resistance and physical protection. In order to improve the economics, the rated power was increased from the 600 MW(e) of the KALIMER-600 to 1200 MW(e) [2, 3]. Breakeven cores loaded with metallic fuels do not have blankets in order to strengthen the proliferation resistance and employ a safety grade residual heat removal system, the PDRC, which operates passively by natural circulation. Table 1 shows the key design parameters of the advanced SFR being developed at KAERI.

Two types of conceptual core design, and breakeven and TRU burner cores were developed. The breakeven core is a reference concept for the 1200 MW(e)

TABLE 1. KEY DESIGN PARAMETERS OF THE ADVANCED SFR

Overall		**PHTS**	
Net plant power (MW(e))	1200.0	Reactor core I/O temperature (°C)	390/545
Core power (MW(th))	3046.4	Total PHTS flow rate (kg/s)	15 455.4
Gross plant efficiency (%)	41.9	Primary pump type	Centrifugal
Net plant efficiency (%)	39.4	Number of primary pumps	4
Reactor	Pool type		
Number of IHTS loops	2	**IHTS**	
Safety decay heat removal	PDRC	IHX I/O temperature (°C)	325/528
Seismic design	Seismic isolation bearing	IHTS total flow rate (kg/s)	11 777.7
		IHTS pump type	Centrifugal
		Total number of IHXs	4
CORE			
Metal alloy fuel form	U-TRU-10%Zr	**SGS**	
Conversion ratio	1.0	Steam flow rate (kg/s)	1326.6
		Steam temperature (°C)	503.0
		Steam pressure (MPa)	16.5
		Number of SGs	2

Note: PHTS: primary heat transport system; IHTS: intermediate heat transport system; IHX: intermediate heat exchanger; SGS: steam generator system.



(a) 1200 MW(e) breakeven core *(b) 600 MW(e) TRU burner core*

FIG. 4. Layout of cores.

advanced SFR. According to the current study [4], the TRU burning rate increases linearly with the rated core powers from 600 MW(e) to 1200 MW(e). Considering (i) the realistic size of the SFR demonstration reactor planned to be constructed by 2028 in accordance with the long term R&D plan, and (ii) the availability of a KALIMER-600 reactor system design, a TRU burner of 600 MW(e) was selected. Figure 4 and Table 2 show the layout and key design parameters of the two cores, respectively.

The heat transport system comprises a PHTS, an IHTS, an SGS and a residual heat removal system. The heat transport system was established through trade studies in order to enhance the safety and to improve the economics and performance of the KALIMER-600 design. From the study, the heat transport system of the advanced SFR has design features such as two IHTS loops, a Rankine cycle energy conversion system, two double-wall straight tube type SGs and a passive decay heat removal system, as shown in Fig. 5.

TABLE 2. KEY DESIGN PARAMETERS OF THE CORES

Core design parameters	Breakeven core	TRU burner core
Power (MW(e))	1200	600
Core height (cm)	80	89
No. of fuel regions	2	3
Cycle length (effective full power month)	18	11
Charged TRU enrichment (inner, middle, outer cores, wt%)	13.16/ - /16.79	30.0
Conversion ratio (fissile/TRU)	1.0/-	0.74/0.57
Sodium void reactivity (EOEC, $)	7.25	7.50

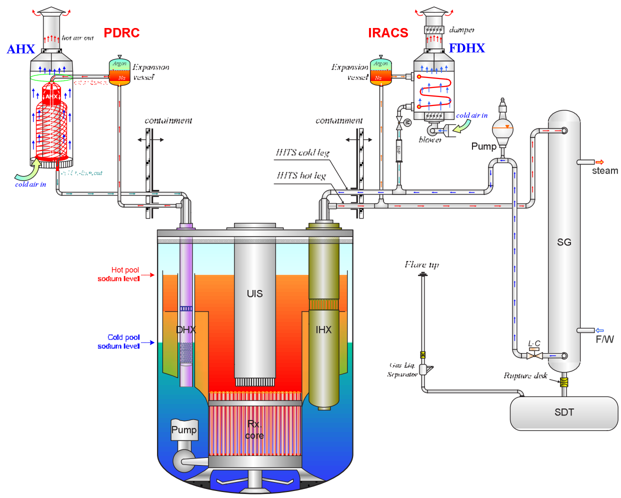


FIG. 5. Configuration of the heat transport system.

The passive decay heat removal circuit (PDRC) consists of four independent loops, and each loop is equipped with a single sodium-to-sodium decay heat exchanger (DHX), a single sodium-to-air heat exchanger (AHX), and the piping connecting the DHX with the AHX. During normal plant operation, the DHX is partially dipped into the cold pool sodium in order to prevent unexpected freezing of the PDRC loop sodium. Under accident conditions, such as a total loss of normal heat sink, the level of the cold pool is raised to that of the hot pool because of the loss of head difference between the hot and cold pools, similar to the PHTS pump trip following the reactor shutdown, as depicted in Fig. 6.

After reactor shutdown, the level of sodium increases from the expansion of primary sodium due to accumulation of reactor core decay heat. If the sodium level increases higher than the slots in the DHX, the hot pool sodium overflows into the shell-side of the DHX. As the sodium flow rate through the shell-side of the DHX increases, the heat transfer rate through the DHX increases due to the enhancement of convective heat transfer. The heat transferred to the PDRC is finally dissipated into the atmosphere through the AHX by natural circulation in the PDRC loop.

In order to secure the economic competitiveness of an SFR compared with a PWR, several concepts were implemented in the mechanical structural design without losing the reactor safety level. Figure 7 shows the reactor internals and component arrangement in the reactor vessel. The material of the reactor vessel and the internal structure is a type 316 stainless steel. The outer diameter of the

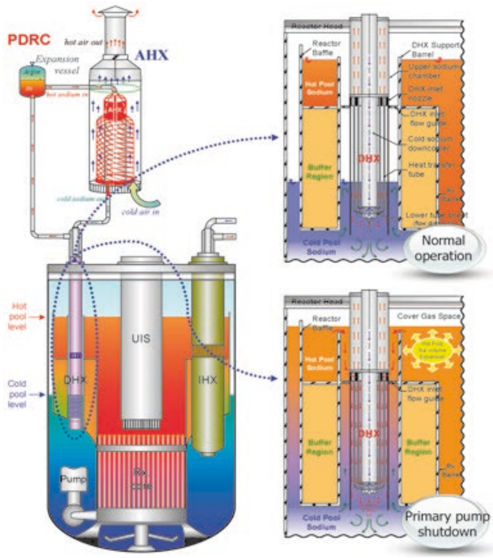


FIG. 6. PDRC configuration and decay heat removal process.

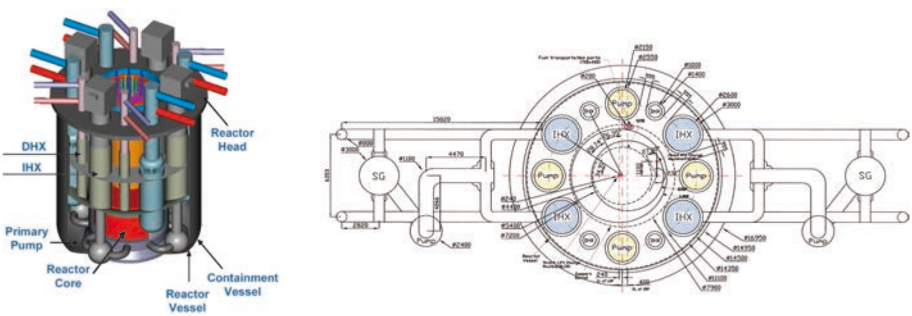


FIG. 7. Preliminary NSSS arrangements of two loop system.

reactor vessel is 14.5 m, which is a very compact size compared with other designs. The primary system consists of four sets of primary pumps, IHX and DHX in the reactor vessel. Each intermediate heat transport loop has a mechanical type pump and an SG connected by large diameter pipes. The piping material is a Mod.9Cr–1Mo alloy, which can allow shortening the piping length to about 60 m compared with stainless steel, because of its higher mechanical strength and lower thermal expansion. The piping diameters for the hot and cold legs are 80 cm and 110 cm, respectively.

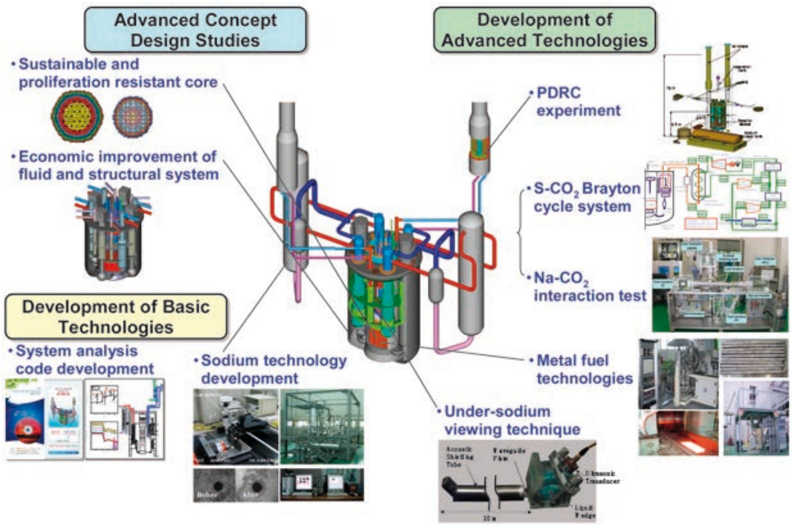


FIG. 8. R&D activities for the advanced SFR.

3.2. R&D activities for the advanced SFR

Various R&D activities have been performed in order to support the development of advanced design concepts and features which will better meet the Generation IV technology goals on sustainability, safety and reliability, economics, proliferation resistance and physical protection, as shown in Fig. 8. These activities include the PDRC experiment, the conceptual design of the supercritical carbon dioxide (S–CO_2) Brayton cycle system, the Na–CO_2 interaction test, under sodium viewing technique, sodium technologies, development of codes and validation, and metal fuel.

4. PYROPROCESS TECHNOLOGY DEVELOPMENT

Pyroprocessing is one of the promising technologies used to treat spent fuel and to reduce its volume [5–7]. It mitigates a repository burden by the separation of uranium from spent fuel and shortens a repository management period by transmuting TRUs. The pyroprocessing technology listed in the long term development plan includes an electrolytic reduction system of PWR spent fuel, a high throughput electrorefining system, an electrowinning system for TRU recovery, waste salt regeneration and solidification, and system engineering technology development.

4.1. Pyroprocessing technology

The oxide reduction process was widely studied as a front end of the pyroprocess and the programme in KAERI was launched in 1997. At first, the lithium based reduction process for spent fuel was developed (1997–2000). The process, with a several kgU/batch capacity, was installed and evaluated between 2001 and 2003. After successfully completing the process with a 5 kgU/batch, KAERI developed a 20 kgU/batch process and obtained more than a 99% reduction yield of uranium from the inactive test. KAERI adopted a new electrochemical reduction process in March 2002 for the purpose of integrating the two existing processes: a lithium based reduction process and a lithium based electrowinning process. The experimental results achieved by KAERI showed that the new process is much improved, compared with the conventional lithium reduction process, in terms of simplicity, safety and throughput. In 2005, KAERI installed an electrolytic reducer in the advanced spent fuel conditioning process facility and performed seven inactive tests with fresh U_3O_8 and SIMFUEL. A LiCl–Li_2O molten salt was used as an electrolyte and the electrochemical reactions were tested. Since the test result showed no LiCl decomposition, it ensured the stability of the LiCl molten salt. Each run was performed with ~10 kg of fresh U_3O_8 or SIMFUEL. The extent of fuel reduction and the distribution of stable fission products between the salt and fuel phases were determined. A reactor model was developed to assess scale-up issues for a high throughput electrolytic reducer. During these inactive tests, a reduction yield of more than 99% and an anode current density of more than a 100 mA/cm^2 were obtained. In 2007, two bench scale (~50 gHM/batch) electrolytic reduction runs using a KAERI cathode basket were completed with spent LWR fuel in a hot cell at Idaho National Laboratory's Materials and Fuels Complex, securing data on the effects of fission products.

In 2008, the design and construction of a new electrolytic reduction system (20 kgU/batch) equipped with a metal cathode basket, which can be linked to an electrorefining process, was completed and a performance test is now under way (Fig. 9) [8]. The technology to suppress the vapourization of molten salts and enable reuse of the molten salts was verified. Current density on the anode was increased from ~100 mA/cm^2 (old electrolytic reducer) to ~500 mA/cm^2 (new electrolytic reducer) enabling high speed electrolytic reduction.

The study on electrorefining technology started in 1997. At the beginning of the programme, basic experiments on the thermodynamic properties and chemical characterization of unit processes were performed. On the basis of the fundamental study, electrorefiners of 100 gU/batch and 1 kgU/batch were successfully developed in 2003 and 2006, respectively. The efficiency of the process is one of the requirements for the treatment of a large volume of uranium

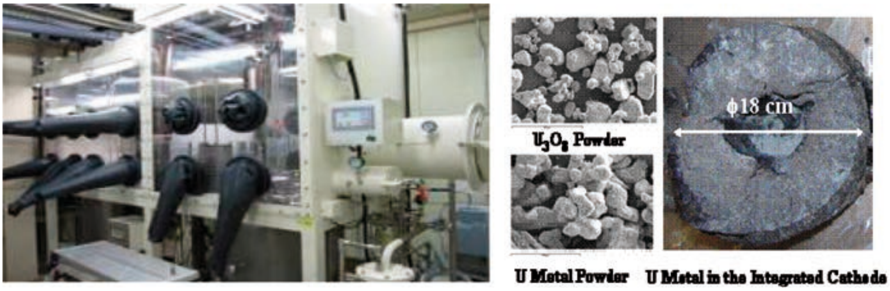


FIG. 9. Laboratory scale electrochemcial reduction system.

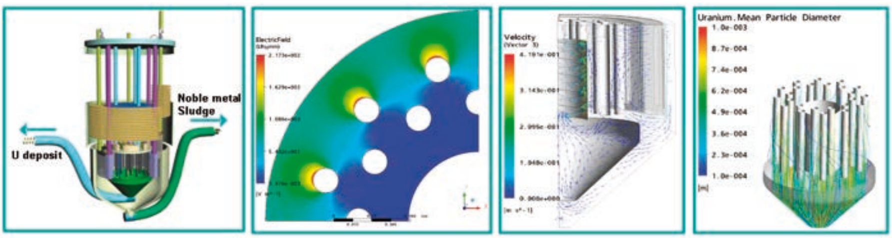


FIG. 10. Computational fluid dynamic and electrical field analysis of the electrorefiner.

in an integrated area. Because of this, many attempts have been made to enhance the throughput of an electrorefiner. The main concepts for enhancing their throughput are to increase the cathode area and decrease the physical distance between the electrodes, as well as to operate the electrodeposition process continuously without any interruptions. The conventional batch type electrorefiner, however, uses an iron based cathode and thus requires a mechanical scraping process and/or a stripping process in order to collect the electrolytically deposited uranium at the cathode into a basket.

Such a requirement makes it impossible for these electrorefiners to provide continuous operation and, thus, obtain a large quantity of products in a limited time. To overcome the above drawbacks of the existing devices, KAERI has been developing a new electrorefiner concept by introducing a graphite cathode, which has a self-scraping characteristic for uranium deposits, and computational fluid dynamic and electrical field analyses were performed, as shown in Fig. 10 [9]. The self-scraping of the graphite cathode occurs first as the uranium–graphite intercalation compounds are formed on the surface of the cathode. When the gravitational force of the deposited uranium dendrite exceeds the bonding strength of the elongated graphite's outermost layer, the self-scraping mechanism occurs by itself. With this self-scraping mechanism, the KAERI electrorefiner

FIG. 11. Continuous high throughput electrorefiner installed at KAERI.

does not require a mechanical scraping process. In addition, a stripping process is also unnecessary because there would be no residuals stuck to the surface of the cathode after the uranium metal is separated. As a result, the electrodeposition process could be continued as long as the material is supplied to the anode basket. The electrorefiner employs screw-type conveyors to extract the uranium deposit and transition metal impurities separately from a high temperature molten salt and installs two separate collecting basins for the electrodeposited uranium and the undissolved transition metal particles. Recently, KAERI installed a continuous high throughput electrorefiner with a capacity of 20 kgU/d and performance testing is under way, as shown in Fig. 11.

Electrowinning technology which recovers TRUs using a liquid cadmium cathode (LCC) is recognized as a proliferation resistant process since the electrodeposition potentials of the actinides have nearly the same values. According to research results from other countries, the inhibition of the growth of uranium dendrites has been considered a key factor in the improvement of the LCC performance because the uranium dendrites growing at the interface between molten salt and cadmium prevents the electrodeposition of TRU elements [10]. KAERI has manufactured a laboratory scale experimental apparatus (Fig. 12) and LCC performance tests are being conducted for the development of an innovative LCC structure and operating method to remove the uranium dendrites on the surface of the cadmium cathode.

In order to minimize the amount of waste generated from the pyroprocessing system and the content of actinides to be disposed of, it is necessary to establish an effective residual actinide recovery method for treating the spent salt remaining after an application of the electrowinning step prior to removal of all the fission products in the waste salt treatment step (Fig. 13). Various methods such as reductive extraction, electrodeposition, or oxidation, etc., are considered as a candidate residual actinide recovery technology in other

FIG. 12. Experimental apparatus for the electrowinning process.

countries. Recently, KAERI developed a hybrid concept using an LCC and an oxidant based on the results of the thermodynamic approach. It consists of, first, electrolysis using an LCC to collect all the residual actinides and some lanthanide fission products to reduce the concentration of actinides in molten salt. Secondly, there is a selective recovery of parts of the accompanied lanthanide fission products by oxidation (or chlorination) from a Cd alloy with fission product, U, and TRU. Presently, experimental works are being carried out to confirm the applicability of this innovative residual actinide recovery technique.

During the pyroprocessing of LWR spent oxide fuel, two different types of waste salt are expected to be generated: (i) a LiCl waste salt containing alkali and alkaline-earth (Group I/II) fission products from an electrolytic reduction process, and (ii) a LiCl–KCl eutectic waste salt containing rare earth fission products from an electrorefining process. Since these waste salts are radioactive, heat generative and highly soluble in water, they must be fabricated into durable waste forms that are compatible with the environment existing within a geological repository for a long time. Current technology for disposing of waste salts from the pyroprocess is non-selective total incorporation of the salts in a zeolite matrix in a ceramic waste form, which results in a significant increase in the final waste volume for disposal.

KAERI has two key R&D concepts in developing innovative waste salt treatment technology [11]. The first one is the minimization of waste salt generation by the removal of fission products followed by recycling of the cleaned salt into the main pyroprocesses. The second one is the increase in safety during interim storage or final disposal by the fabrication of high integrity final waste forms. To meet these purposes, KAERI has developed various kinds of fission product removal and waste solidification technologies such as melt crystallization, oxidative precipitation and superabsorbent polymer solidification. The performance was found to be successful in small scale equipments. Scale up

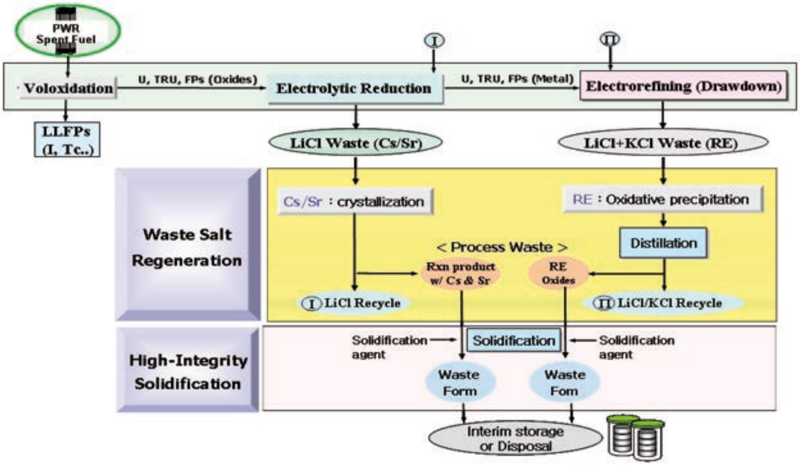


FIG. 13. KAERI's approach to the effective management of waste salts.

FIG. 14. Bird's-eye view of the PRIDE facility.

of the salt regeneration and solidification units and verification of the performance in the scaled up processes will be done in a stepwise manner according to KAERI's long term R&D plan, which is financially supported by the national long term nuclear R&D programme.

4.2. Pyrosystem engineering technology

The PRIDE facility as shown Fig. 14 is a three storey building with a large argon cell on the second floor. The inner dimension of the argon cell is 40 m × 4.8 m × 6.4 m. The PRIDE facility has been designed with a stringent inert atmosphere control and fully remote operation concepts. This facility will be used

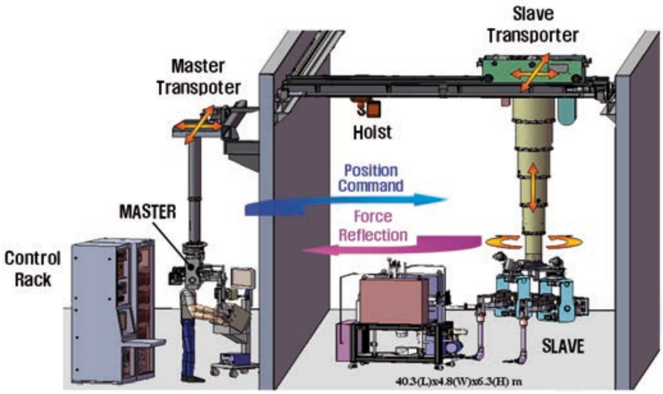


FIG. 15. The bridge transported dual arm servomanipulator system for the PRIDE facility.

to evaluate the integrated pyroprocess concepts and produce reliable data for scale up issues.

Remote operation and maintenance of the facility are some of the key technologies required. In this regard, remote handling systems such as a bridge transported dual arm servomanipulator and a simulator are being developed for the PRIDE facility application, as shown in Fig. 15. They will provide efficient tools for remote operation and maintenance work in pyroprocessing technology development at the PRIDE facility, thereby reducing an operator's burdens, improving the performance of the process equipment operation and making the PRIDE facility more functional.

The safeguards system development is also one of the most important research items for the succesful development of the pyroprocess. In the safeguards aspect, KAERI has researched the development of nuclear material accounting systems for a pyroprocessing facility. The major accounting technique is the curium ratio approach, in which a neutron counting system should be able to measure pyroprocess materials with high accuracy. For front end pyroprocess material accounting, KAERI has developed the advanced spent fuel conditioning process safeguards neutron counter (ASNC), as shown in Fig. 16, which has special features such as hot cell operation and remote maintenance capabilities [12]. The test and calibration results using spent fuel samples showed excellent performance [13]. Further development of non-destructive assay technologies has been undertaken to complete the nuclear material accounting system for PRIDE and an engineering scale process facility.

KAERI is also involved in a joint R&D programme with the IAEA to develop efficient safeguarding systems for future pyroprocessing facilities. All of these innovative concepts will be incorporated into an engineering scale PRIDE

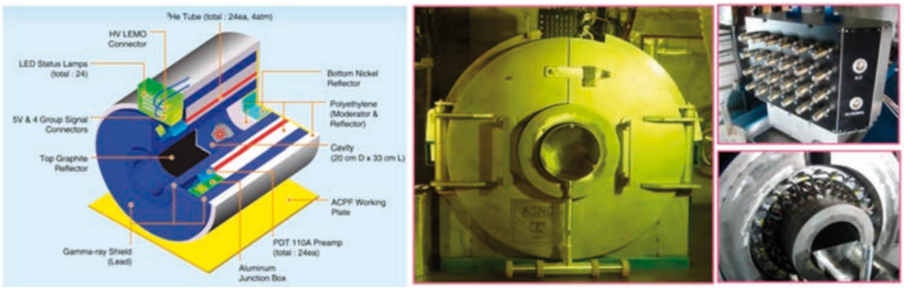


FIG. 16. Diagram of the ASNC components and the ASNC in a hot cell.

facility scheduled to be built by 2011 to verify its technical and economic viability.

5. SUMMARY

The long term advanced SFR and pyroprocess R&D plan authorized by the KAEC will be carried out. The milestones of this long term plan are the construction of a pilot scale facility (KAPF) by 2025 and the construction of an advanced SFR demonstration plant by 2028. Currently, a detailed implementation plan is being developed.

On the basis of the experiences gained during the development of conceptual designs for KALIMER, KAERI is developing key technologies for an advanced SFR. There are three categories of activities under way: (i) advanced concept design studies, (ii) development of the advanced SFR technologies necessary for its commercialization, and (iii) development of basic technologies.

Pyroprocessing is a useful technology for recycling spent fuel. However, this technology has only been studied at a laboratory scale to date. For practical use of this technology, knowledge of a scale up is essential. In 2011, PRIDE will be constructed and used for testing the integrity of the unit process, the adaptability of remote operation and the safeguardability on an engineering scale. PRIDE will be open for international collaboration.

ACKNOWLEDGEMENT

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NATIONAL AND INTERNATIONAL FAST REACTOR PROGRAMMES

(Plenary Session 2)

Chairpersons

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THE PROGRAMME FOR FAST REACTOR DEVELOPMENT IN THE RUSSIAN FEDERATION

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Abstract

The paper highlights the status and perspectives on the development of nuclear energy based on fast reactor and closed fuel cycle technologies in the Russian Federation. Information is presented on the new Federal Target Programme "Nuclear Power Technologies of a New Generation for the Period 2010–2015 and the Outlook to 2020".

1. INTRODUCTION

Currently, nuclear energy represents a significant share in the energy supply of the Russian Federation, producing some 16% of total electricity. Growing ecological and resource challenges related to fossil fuel consumption, which are predominant nowadays, require finding ways of achieving a considerable extension of nuclear energy's role.

Nuclear power in the Russian Federation and worldwide is currently based on technologies which employ thermal reactors and a water coolant. Modern nuclear power plants are acceptably safe, ecologically attractive, and if the delayed problems posed by spent nuclear fuel (SNF) are disregarded, are competitive in terms of electrical energy production.

However, current technologies cannot provide the foundation for large scale nuclear energy for two main reasons:

(1) Low efficiency in the use of natural uranium (less than 1%);
(2) Storage of SNF, with quantities steadily increasing worldwide.

The scientific, design and technological research resources for finding the ways to solve these problems have been under way for more than 50 years. The development of the closed fuel cycle technological package employing a fast reactor as a key technological component has been one of the most promising areas. The technological package includes fabrication of mixed uranium–plutonium fuel, reprocessing of SNF and recycling of separated nuclear materials in fast reactors. It is considered in the Russian Federation that, if brought to the commercial level, these are the technologies that will form the basis of the new technological platform for large scale nuclear power.

2. CURRENT STATUS AND MEDIUM TERM PROSPECTS FOR NUCLEAR ENERGY IN THE RUSSIAN FEDERATION

There are currently 31 power units in operation in 10 nuclear power plants in the Russian Federation; their total installed power being equal to 23.2 GW(e). Their energy production in 2008 was 162.3 billion kW· h, or 16% of total electricity production. The average load factor of the plants reached 79.5% and plant operation is characterized by high reliability and safety. In recent years, no incidents resulting in radiological consequences nor anomalies exceeding level zero of the International Nuclear and Radiological Event Scale have occurred.

A recently approved new strategy for energy sector development in the Russian Federation is based on the assumption that electricity production remains the main body of energy economy. The total capacity of power plants in the country is expected to increase from 225 GW(e) to 355–445 GW(e) by 2030. Nuclear power plant capacity is expected to increase to 37–41 GW(e) by 2020 and up to 52–62 GW(e) by 2030, with the nuclear share in electricity production increasing by up to 20%.

Development of nuclear energy is one of five top priority areas in the recently announced President's plan for technological renovation of national economics. In the nuclear energy area there are two projects under consideration. The first project is optimization of WWER technology; the second is creation of a closed fuel cycle infrastructure that incorporates fast reactors.

3. STATUS OF FAST REACTOR AND CLOSED NUCLEAR FUEL CYCLE TECHNOLOGIES

A unique programme of sodium cooled fast reactor (SFR) development had been initiated in the former USSR by commissioning of the experimental reactor BR-5 at the IPPE (Obninsk) in 1959. Experience gained in the course of development, construction and operation of the experimental BR-5(10) reactor, as well as with the BOR-60 (Dimitrovgrad) research reactor, and the BN-350 (Shevchenko) and BN-600 (Zarechny) semi-industrial reactors allowed the start of designing and building the BN-800 high capacity commercial reactor. Some detailed information on their status is given below.

BOR-60. Experimental reactor BOR-60, commissioned in 1969, is currently used for materials and various components of fast reactor tests, as well as for isotopes, heat and electricity production. Reactor rated power is 60 MW(th). The work under way is aimed at extending its lifetime from 2010 to 2015.

BN-600. Number 3 power unit of the Beloyarsk nuclear power plant, which has the world's largest fast reactor (BN-600) has been in operation for 29 years. The unit's operation over all this period is characterized by consistent high reliability, safety and cost effectiveness. The problems related to sodium technology were successfully resolved within the first 6–7 years following commissioning. An average load factor for 2008 was 77.5%, and for the whole period of power unit commercial operation (1982–2008) it was 75.7%. Despite the BN-600 reactor being the first high power integrated reactor, this key parameter is similar to that of domestic commercial LWRs. The BN-600 power unit design lifetime (30 years) will expire in April 2010. Work is currently under way to extend the power unit's lifetime to 2025.

BN-800. The development of BN-800 was initiated immediately after the completion of work on the BN-600 reactor in 1980. A small series of four reactors was planned in the former USSR for construction at the Beloyarsk nuclear power plant and in the southern Urals. However, owing to several reasons, including the Chernobyl accident and the economic recession, the construction of the first two BN-800 reactors initiated at these sites was postponed.

Nevertheless, the work on BN-800 was continued. These efforts were aimed at safety upgrades and improvement of economics. The work in this direction performed in 1990 was considered successful. Modernization of the BN-800 design enabled the electrical output to be increased from 800 MW to 880 MW (while maintaining the same thermal output of the reactor, i.e. 2100 MW) and the power unit lifetime to be increased from 30 to 40 years and, therefore, the technical and economic performance of the power unit to be improved. In 1997, a licence for the resumption of BN-800 construction at the Beloyarsk nuclear power plant was obtained. It was the first licence to be granted for nuclear power plant construction in the Russian Federation since the Chernobyl accident.

In 2006, the Government of the Russian Federation approved the Federal Target Programme “Development of the Nuclear Energy Industrial Complex in the Russian Federation for the Period up to 2015”. Along with the construction of WWER-1000 reactor power units, the programme envisages construction of BN-800 units. The power unit with the BN-800 reactor is currently at the construction stage, with commissioning expected in 2014.

The most important task allotted to the BN-800 reactor concerns mastering of the closed fuel cycle technologies. Other, equally essential tasks are preservation of knowledge, technologies and skills in the area of SFRs and the putting into operation of an additional source of electricity in the Urals region experiencing shortages of energy.

SVBR-100. In addition to the direction of SFRs, the country has gained 40 years experience in the development and operation of reactors with lead–bismuth coolant on nuclear submarines. A conceptual design for the SVBR-100 reactor (lead–bismuth fast reactor of 100 MW equivalent electrical power) was developed on the basis of this experience. The supposed niche for the commercial application of the SVBR-100 reactor for domestic use is in regional power engineering, where the predominant share of electrical power is generated at cogeneration plants that have to be located in the vicinity of towns.

BREST. Research work is currently being conducted on lead cooled reactor technology with uranium–plutonium mononitride fuel. The conceptual design of the BREST reactor envisages an on-site arrangement of the closed nuclear fuel cycle with an application for dry pyrochemical fuel reprocessing technologies.

As to the development of the closed fuel cycle and radioactive waste management technologies in the country, the current status is as follows:

(a) The technology for the aqueous chemical reprocessing of SNF from thermal reactors with uranium and plutonium separation and vitrification of high level radwaste has been demonstrated at an industrial level (RT-1 plant).
(b) The pellet and VIPAC technologies for the MOX fuel of SFRs have been demonstrated at an experimental level.
(c) The R&D on the development of advanced fuel cycle technologies is under way (nitride fuel, dry reprocessing of SNF, transmutation of minor actinides in fast reactors and uranium–thorium cycle).

4. LONG TERM PROSPECTS FOR NUCLEAR ENERGY IN THE RUSSIAN FEDERATION

Assurance of the energy security of the country requires a considerable increase in nuclear power. Currently, it is difficult to foresee a specific level of total nuclear power plant capacity being reached. However, in order to assure a credible role for domestic nuclear energy in the electrical power industry, the level of nuclear capacities should be at least 100 GW(e) by the middle of the century.

In our understanding, this large scale nuclear energy infrastructure would be possible under the following conditions:

(a) Meaningful share of safe and cost effective fast breeder reactors in the nuclear energy infrastructure;
(b) Closure of the nuclear energy fuel cycle with multiple recycling of uranium and plutonium in fast reactors, with optimal ways of managing the recycling of minor actinides and fission products;
(c) Availability of proliferation resistant exportable fast and thermal reactors;
(d) Establishment of international centres for rendering nuclear fuel cycle services.

It should be noted that all of the above mentioned technological and institutional solutions for the large scale nuclear energy infrastructure still require development. In this case, there emerges the problem of optimization of a transition strategy while taking into account specific features for development of nuclear energy:

(a) The whole life cycle of a nuclear energy system (starting from scientific research, creation of a demonstration plant and operation of a commercial nuclear power plant up to its decommissioning and solution of the radwaste problem) may last over 100 years;
(b) The need for considerable scientific and financial resources to be allocated over a very long time period (several decades) in order to master technologies at the commercial level;
(c) The application of dual use technologies and materials in nuclear energy.

Because of the above features, commercial industry has no practical interest in the development of the new nuclear energy technologies. On the other hand, the above features of necessity require a significant increase in the State's role in the development and stewardship of nuclear power complexes. It should also be taken into account that the commercial technologies of thermal reactors and the open fuel cycle currently used at the industrial level were basically developed at the expense of the State budget, including funds for national defence programmes.

Taking into account financial and technological risks, the transition strategy should provide the possibility for a step-by-step correction of adopted approaches as external economic and technological characteristics are revised over time, keeping to the main vector of nuclear energy development.

For the purpose of realizing the large scale domestic development of nuclear energy during the first half of this century, it seems necessary to demonstrate over the medium term (before 2030) the technological package of key elements for the new technological platform infrastructure, including:

(a) A small series of commercial power units with fast reactors (about 10–15 GW(e) power);
(b) A MOX fuel fabrication facility to produce fuel for the small series of fast reactors;
(c) An LWR SNF reprocessing plant employing advanced aqueous technologies;
(d) Development of fast reactor SNF reprocessing technologies.

Creation of such an infrastructure would demonstrate the possibility for the comprehensive solution to problems related to current nuclear energy, i.e. SNF accumulation, low effectiveness of use of natural resources and non-proliferation assurance. In particular, the timely reprocessing of WWER SNF would make it possible not only to abandon putting into service new storage facilities for SNF, but also to minimize accumulation of highly radiotoxic ^{241}Am in the course of the long term storage of SNF and, thereby, simplify the solution to the problem of final disposal of high level waste.

Reprocessing of SNF from thermal reactors would allow starting regenerated uranium recycling in thermal reactors, which would reduce consumption of natural uranium in the present system by 10–20%. However, the principal benefit in the area of fuel supply will manifest itself after 2030, i.e. when commercial fast reactors could be intensively deployed.

5. FEDERAL TARGET PROGRAMME ON NUCLEAR POWER TECHNOLOGIES

With a view to creating and demonstrating key elements of the new technological platform by 2020, the Federal Target Programme (FTP) “Nuclear Power Technologies of a New Generation for the Period 2010–2015 and the Outlook to 2020” was prepared and submitted for approval to the Government. To realize the goal set, the FTP suggests activities in the following areas:

(a) Development of advanced fast reactor technologies;
(b) Design and construction of new experimental facilities and equipment, and upgrading and development of experimental and test facilities for justification of the new fast reactor technologies;
(c) Development of technologies for advanced fuel fabrication for the new generation reactors;
(d) Development of reprocessing technologies.

In the framework of the FTP, the method of simultaneous development of several reactor technologies is accepted as a basic approach, namely, fast reactor with sodium coolant (SFR), fast reactor with lead coolant (BREST) and fast reactor with lead–bismuth coolant (SVBR) and related fuel cycles.

The FTP realization is planned in two phases:

(1) In the first phase, the following activities should be included among those planned for the period 2010–2014:
   — Development of basic designs of prototype fast reactors, as well as related technologies for the closed fuel cycle;
   — Completion of design and commissioning of MOX fuel fabrication facility (BN-800);

— Development of detailed design for the construction of a multipurpose research fast reactor (MBIR) using sodium coolant.

(2) In the second phase (2015–2020), the following main activities will be fulfilled:
- — Construction of prototypes of the BREST and SVBR reactors;
- — Commissioning of upgraded critical facilities;
- — Construction of the pilot plant for fabrication of dense fuel for fast reactors;
- — Construction of the prototype pyrochemical complex;
- — Construction, reconstruction, technical re-equipment and commissioning of experimental facilities for justification of the new technological platform for nuclear power, including construction of the MBIR.

The FTP is aimed at the scientific and technological support of those innovative reactor technologies which still require experimental confirmation of their feasibility at the level of prototypes (BREST and SVBR). For SFRs, which have proven technically viable in the case of examples such as BOR-60, BN-350 and BN-600, and which are the most advanced of all the reactor types listed above, the tasks are set for a further improvement of the technical, economic and safety characteristics to the level that would meet the requirements of the Generation IV reactors. This includes construction of a BN-800 reactor with MOX fuel to demonstrate closure of the fuel cycle, as well as design and construction of a new advanced commercial SFR (BN-C).

For the proper management of different activities within the FTP, creation of a coordination centre is envisaged.

6. CONCLUDING REMARKS

The problem of energy supply for sustainable development in the Russian Federation can be resolved by means of the creation and step-by-step deployment of large scale nuclear power, based on a new technological platform for the closed fuel cycle with fast reactors.

International cooperation plays an important role in this process. One step in this direction is the leading participation of the Russian Federation in the International Project for Innovative Reactors and Nuclear Fuel Cycles (INPRO), which successfully proceeds under the aegis of the IAEA. Another step involves participation of the Russian Federation in the Framework Agreement for International Collaboration on Research and Development of Generation IV Nuclear Energy Systems.

With the FTP on nuclear power technologies being implemented, the Russian Federation will be open to wider international cooperation in the area of fast reactor and closed fuel cycle technologies development. This might include multilateral designing, construction and mutual use of a new multipurpose research fast reactor — the MBIR.

THE US ADVANCED FUEL CYCLE PROGRAMME: OBJECTIVES AND ACCOMPLISHMENTS

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Abstract

For approximately a decade, the United States Department of Energy has been conducting an advanced fuel cycle programme, presently named the Fuel Cycle R&D Program, devoted to lessening both the environmental burden of nuclear energy and the proliferation risk of accumulating used nuclear fuel. Currently, the programme is being redirected towards a science based, goal oriented focus with the objective of deploying successfully demonstrated technology in the 2040–2050 time frame. The present paper reports the key considerations of the science based research approach, the elements of the technical programme and the accomplishments in fast reactor research and development, the goal of which is to improve the primary issues that have inhibited fast reactor introduction in the past, namely, economics and safety.

1. INTRODUCTION AND BACKGROUND

For approximately a decade, the United States Department of Energy has been running advanced fuel cycle research in the current Fuel Cycle R&D Program and its predecessors. A key objective of this research has been improved waste management by lessening both the environmental burden of nuclear energy and the proliferation risk of accumulating used nuclear fuel.

Until recently, this programme was technically focused on achieving an optimized symbiosis between fuel cycle options, on the one hand, and the US geological repository on the other. In previous years, a relatively short term deployment focus was being pursued. On the basis of detailed technical analyses, this focus led to the selection of a limited set of technologies that were expected to meet specific geology related criteria, and which would be based on limited extrapolations of existing technologies.

Recent developments in the United States of America indicate that alternative repository sites will be considered and that Yucca Mountain may not be the choice for final disposal. With delayed geological disposal, advanced fuel cycles could be postponed until mid-century, with increased reliance on temporary storage of used nuclear fuel in the interim.

2. FUEL CYCLE R&D OBJECTIVES AND APPROACH

Consequently, the Fuel Cycle R&D Program is being redirected towards a longer term science based research approach. The work will be conducted with a goal oriented focus, driven by the following three considerations:

(1) The programme is currently examining a broad set of options, including different geological media and transmutation technologies in order to understand their relationships and provide information for later decisions.
(2) The R&D component of the programme is focused on acquiring the basic understanding of key phenomena, defining the relevant challenges and acquiring the basic tools necessary to resolve them.
(3) The timeline of the programme allows for deployment of the successfully demonstrated technology in the 2040–2050 time frame; allowing consideration of technologies that are not yet mature but that might provide significant improvements in performance.

The science based research approach will integrate theory, experiment and high performance modelling and simulation to promote development of the needed technologies. The focus for a science based approach shifts to smaller scale experiments of phenomenological and separate or coupled effects. This approach provides a fundamental understanding of targeted phenomena and data for model development. New and innovative experimental design and novel measurement techniques are anticipated to support improved fidelity and small scale detail.

Theory development is an essential element of the science based approach. This requires first a deep understanding and a database of existing knowledge and theories to identify and explain the key phenomena. In the long term, theory must span quantum mechanics to continuum mechanics in order to explain the behaviour of physical systems. A well-integrated balance between experiments and theory development is required.

The knowledge and data gained under the experimental and theoretical elements of the science based approach will be incorporated into advanced modelling and simulation tools that take advantage of state of the art computing capabilities. Owing to the complex and formal nature of the nuclear licensing process, procedures to demonstrate the validity of the new tools must be clearly addressed.

The technical programme is articulated along the following elements:

(a) A systems integration task that analyses the relationships between technologies and defines requirements to achieve overall system objectives.
(b) A separation research programme that is aimed at understanding the fundamentals of actinide chemistry in order to develop processes that achieve specific separation goals with very low losses.
(c) A better understanding of used nuclear fuel geological repository options and development of better safeguards techniques.
(d) A fuels research programme that is also aimed at a better understanding of the fundamentals of fuel behaviour in order to design minor actinide containing fuels with high burnup capabilities.
(e) A fast reactor research programme aimed at reducing the cost of fast reactors, with increased safety performance. This work is conducted as part of the Generation IV programme in close collaboration with the Fuel Cycle R&D Program.

3. FAST REACTOR R&D ACCOMPLISHMENTS

The role of the reactor in a closed fuel cycle is to utilize materials recovered from spent fuel for both electricity production and fuel cycle management. Recycling of key elements is required to satisfy both waste management and resource extension objectives. A fast spectrum reactor with an associated closed fuel cycle is required to close completely the fuel cycle because practical limitations to extended recycle have been identified for thermal systems.

A variety of fast reactors can be considered for transmutation. Three options being considered in the Generation IV advanced reactor programme are: (i) the sodium cooled fast reactor (SFR), (ii) the lead alloy cooled fast reactor (LFR) and (iii) the gas cooled fast reactor (GFR). Similar transmutation performance for these systems has been demonstrated. The SFR technology was favoured in predecessor programmes because of its maturity for near term application, while the alternative LFR and GFR technologies offer some advantages for high temperature applications.

Current fast reactor R&D is focused on the primary issues that have inhibited fast reactor introduction in the past:

(a) A perception of higher capital costs as compared to conventional LWR technology;

(b) Unique concerns related to alternative coolants (e.g. sodium reactions with air/water, corrosion by lead alloys, component access under liquid metal, decay heat removal with gas).

Thus, the objective of fast reactor R&D is to research and develop advanced technologies that significantly improve both economic and safety performances of fast recycle systems. This outcome requires concurrent efforts on science based R&D for innovative technologies, integration of new features into reactor systems and development of fast reactor recycle fuels.

3.1. R&D on innovative technologies

Because capital investment in reactors is the dominant cost of any nuclear fuel cycle, this work is critical to assure an economically viable closed fuel cycle. To reduce the cost of future fast reactors, a variety of innovative solutions are being researched:

(a) *Advanced modelling and simulation.* Reactor simulation requires modelling diverse, coupled physics, including neutronics, thermal fluid dynamics and structural phenomena. New techniques will exploit modern computational hardware and visualization software. The improved modelling will make reactor design tools more predictive, reducing the reliance on calibration and conservative margins. Improved accuracy and better integration of methods will also promote design optimization. Improved nuclear data is also important for both system optimization and safety assurance. Prioritized high accuracy experiments are conducted for key actinides and materials used to predict key reactor parameters such as criticality, transmutation rates and reactivity feedback coefficients.

(b) *Advanced materials.* Advanced structural materials could improve reactor costs by enabling compact configurations, higher operating temperatures, higher reliability and longer lifetimes. Modern material science techniques are being used to optimize variants of existing alloys for fast reactor applications. Qualification of these materials requires resolution of code and licensing issues and irradiation testing, and initial testing of candidate alloys is being conducted.

(c) *Advanced energy conversion systems.* Refined energy production systems such as a supercritical CO_2 Brayton cycle offer the promise of improved thermal efficiency. Research needs for advanced heat exchangers (e.g. small tube configurations) and compact components are also being pursued with objectives of both reduced cost and high reliability.

(d) *Safety research.* Inherent safety is a key approach for licensing assurance and cost reduction. A wide variety of design features for prevention and mitigation of severe accidents have been proposed for advanced design concepts. The benefit and performance of features such as core restraint, seismic isolation and ‘core catchers’ are being assessed. In addition, the validation of safety analysis methods and advanced techniques with existing data is being aggressively pursued.

3.2. System integration and concept development

Another important aspect of this work is the analysis of diverse fast reactor technology options (e.g. refined coolants, fuels) and system configuration options (e.g. pool, loop, hybrid, elimination of intermediate loop). The assessment of performance impacts can only be compared in a systematic manner by meticulous application of system constraints and performance criteria.

This work guides the other research activities by providing a fundamental understanding of the technical utilization and feasibility of advanced technology options in an integrated reactor system. Favourable applications for innovative features are developed and the cost reduction benefits are evaluated.

3.3. Fast reactor fuels

The objective of fast reactor fuels R&D is to develop transmutation fuels for use in fast reactors with associated closed fuel cycles. This requires the transmutation fuel to cover a wide range of compositions to account for variability of recycle material feeds and flexible fast reactor fuel cycle modes (e.g. burner or converter). To this end, irradiation testing has been conducted on metal, oxide and nitride fuel forms to assess the impact of including minor actinides and other recycle impurities.

To improve the economic performance of fast reactor recycle, research is also conducted to extend fuel burnup and improve fabrication costs. The realization of high burnup requires the development of radiation tolerant fuel forms and core structural materials. As described above, extensive theory and modelling efforts of fuel behaviour and performance are being pursued to understand and to optimize performance of diverse options. With regard to fuel fabrication, advanced technologies are being developed to allow ‘remotized’ operation, minimize losses and waste, and streamline operations.

FAST REACTOR RESEARCH IN EUROPE: THE WAY TOWARDS SUSTAINABILITY
(Summary)

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The European Union (EU) has taken the lead in responding to climate change, announcing far-reaching initiatives ranging from promoting energy efficient light bulbs and cars to new building codes, carbon trading schemes, development of low carbon technologies and greater competition in energy markets.

Nuclear energy remains central to the energy debate in Europe. One third of EU electricity is produced via nuclear fission and eight new reactors are under construction. Traditionally non-nuclear countries are manifesting an interest in building nuclear power plants while the clock is ticking down on Belgium, Germany and the United Kingdom's decision to renew or close existing nuclear infrastructures.

Sustainability in nuclear energy production is ensured in the medium term as a result of the large and diverse uranium resources available in politically stable countries around the world. The quantities available with high probability ensure more than one hundred years of nuclear energy production. This extrapolation depends, however, on the forecast for future nuclear energy production. The use of fast neutron breeder reactors would lead to a much more efficient utilization of the uranium, extending the sustainable energy production to several thousands of years. The presentation will outline the fast reactors of the new generation currently being developed within the Generation IV initiative.

Broad conclusions of the presentation are that:

- There is a growing nuclear renaissance in Europe for good reason.
- Nuclear energy is a green and sustainable option for Europe and indeed the world's energy needs.
- Nuclear energy is a competitive energy that makes economic sense.
- Nuclear fission reactors have a safety and environmental track record that is second to none, yet public misperceptions persist and must be tackled.
- Waste management solutions exist while new developments hold great promise.

- The evolution and promise of nuclear technologies must also be examined against the costs and risks in a balanced approach.
- Research on fast neutron reactors is being strengthened in Europe, under the umbrella of the Generation IV International Forum. European coordination is entrusted to the Joint Research Centre.

THE IAEA PROGRAMME ON FAST REACTORS, RELATED FUELS AND STRUCTURAL MATERIALS TECHNOLOGY*

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

OECD NUCLEAR ENERGY AGENCY ACTIVITIES RELATED TO FAST REACTOR DEVELOPMENT

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Abstract

The OECD Nuclear Energy Agency (NEA), whose role is to assist its member countries to develop, through international cooperation, the scientific and technological bases required for the safe, environmentally friendly and economical use of nuclear energy, conducts work related to fast reactor systems in two areas of activity: one focused on scientific research and technology development needs and one dedicated to strategic and policy issues. Recent, scientifically oriented, fast reactor related activities coordinated by the NEA comprise:

— A coordinated effort to evaluate basic nuclear data needed for the development of fast reactor systems;
— A recently initiated review of Integral Experiments for Minor Actinide Management;
— An ongoing study on Homogeneous versus Heterogeneous Recycle of Transuranic Isotopes in Fast Reactors;
— A comparative analysis of the safety characteristics of sodium cooled fast reactors;
— A series of workshops on Advanced Reactors with Innovative Fuels;
— A series of information exchange meetings on actinide and fission product partitioning and transmutation.

The NEA has also conducted two reviews on issues related to the transition from thermal to fast neutron nuclear systems. One study was devoted to technical issues, including benchmark studies on: (i) the performance of scenario analysis codes, (ii) a regional (European) scenario and (iii) a global transition scenario. The other study emphasized issues of interest to policymakers, such as key parameters affecting the cost–benefit analysis of transitioning, including the size and age of the nuclear reactor fleet, the expected future reliance on nuclear energy, access to uranium resources, domestic nuclear infrastructure and technology development, and radioactive waste management policy in place. The NEA is also an active player in many other international activities related to fast neutron systems, such as the Generation IV International Forum, where the NEA acts as technical secretariat for the project.

1. INTRODUCTION

The OECD Nuclear Energy Agency (NEA) is an international organization established to assist its member countries in developing the scientific, technological and legal bases required for the safe and economical use of nuclear energy for peaceful purposes. Within this mission, the NEA supports studies related to the development of fast reactor systems, covering both technical and strategic issues. Most of this work has been carried out under the auspices of the Nuclear Science Committee and the Nuclear Development Committee. This paper summarizes recent and ongoing NEA activities in the fields of critical and non-critical fast neutron reactor system development.

2. SCIENTIFIC ISSUES

2.1. Nuclear data

2.1.1. Nuclear data needs for advanced reactors

Since 1989, the NEA has been organizing worldwide cooperation between the major nuclear data evaluation projects, including a review of experimental data needed to improve the quality and completeness of evaluated data libraries (e.g. ENDF, JENDL, JEFF). The work within the nuclear data evaluation cooperation is organized into subgroups, one subgroup for each study.

A recent study [1] focused on the development of a systematic approach to define data needs for advanced reactor systems and to make a comprehensive study of such needs for Generation IV type reactors. A comprehensive sensitivity and uncertainty study was performed to evaluate the impact of neutron cross-section uncertainty on the most significant integral parameters related to the reactor core and the fuel cycle of a wide range of innovative reactor systems.

A compilation of preliminary 'design target accuracies' was undertaken and a target accuracy assessment was performed to provide a quantitative evaluation of nuclear data improvement requirements by isotope, nuclear reaction and energy range, in order to meet the design target accuracies. First priorities were formulated on the basis of the common needs for fast reactors, as well as thermal systems.

This study is being followed up by a new study on methods and issues of the combined use of integral experiments and covariance data, with the objective of recommending a set of best and consistent practices in order to improve evaluated nuclear data files.

2.1.2. Integral experiments for minor actinide management

The establishment of a reliable and economical fuel cycle, including safe management of the radioactive waste, is inevitable in pursuing a sustainable utilization of nuclear fission energy. In this context, minor actinides such as neptunium, americium and curium in the spent fuel should be appropriately managed.

Though various concepts of minor actinide transmutation have been studied, the performance of these concepts is still uncertain due to insufficient knowledge of the accuracy of the minor actinide nuclear data, which are crucial for the detailed design of the transmutation systems, as well as for the accurate prediction of spent fuel composition. Compared with the major actinides, integral experimental data on the minor actinides are scarce due to the restrictions and difficulties in material handling, sample preparation and post-treatment technology.

The NEA has, therefore, recently decided to launch a study [2] to review existing integral experiments for validating minor actinide nuclear data with the aim of recommending additional integral experiments needed for validating minor actinide nuclear data and investigating the possibility of establishing an international framework for promoting integral experiments for minor actinide management.

2.2. Structural materials

Considering the importance of materials in the development of advanced reactor concepts, such as many of the Generation IV concepts where materials will operate at higher temperatures and experience higher radiation damage, the NEA has recently started three activities devoted to structural nuclear materials.

One of these activities is conducted by an expert group [3], which will perform comparative studies to support the development, selection and characterization of innovative structural materials that can be used in advanced nuclear fuel cycles under extreme conditions, such as high temperature, high dose rate, corrosive chemical environment and long service lifetime. The expert group will provide a state of the art assessment of specific areas to be considered as priority areas of research, identify the areas where experimental protocols and standards are needed and where the share of available experimental installations could be possible, identify existing databases, and organize a series of workshops on Structure Materials for Innovative Nuclear Systems.

The objective of these workshops is to exchange information on structural materials research issues and to discuss ongoing programmes, both experimental and in the field of advanced modelling. The first workshop was organized in 2007 in Germany [4], covering technical sessions on the materials for very high temperature reactors, materials for metal cooled reactors, materials for water cooled reactors and a session on multiscale modelling. The second workshop [5] will be held from 31 August to 3 September 2010 in Daejeon, Republic of Korea, and will be hosted by the Korea Atomic Energy Research Institute. The workshop will cover fundamental studies, modelling and experiments on innovative structural materials, including cladding materials for a range of advanced nuclear systems, such as thermal and fast systems, subcritical systems and fusion systems.

The NEA has also established an expert group on structural materials modelling [6] under the Working Party on Multi-scale Modelling of Fuels and Structural Materials for Nuclear Systems. The objective of the expert group is to provide a critical review of the state of the art with respect to the use of a multiscale modelling approach to describe the changes induced by irradiation in structural nuclear materials. To this end, the expert group will review significant examples of the multiscale modelling approach to structural nuclear material applications. It will also identify key problems that should be addressed as priorities towards attaining the goal of developing integrated multiscale modelling frameworks of use in structural nuclear materials applications.

2.3. Fuels

2.3.1. Recycling transuranics in fast reactors

Different approaches for recycling transuranics in fast reactors have been proposed. The two main methods considered are homogeneous and heterogeneous recycling. In homogeneous recycling, the unseparated transuranics are mixed into the fast reactor fuel, independent of fuel form (e.g. oxide or metal) and of reactor coolant type (e.g. sodium, lead, gas). An alternative to homogeneous recycle in fast reactors could be to separate out the less radioactive component of the spent nuclear fuel (e.g. plutonium or neptunium and plutonium in combination) in order to make driver fuels and manage the remaining minor actinides (primarily americium, curium and possibly neptunium) in target fuels/assemblies. Consequently, the driver and target fuels can be managed separately in the fuel cycle. This separate management and recycling of the plutonium driver and the minor actinide target fuels is termed heterogeneous recycling.

Potential advantages and disadvantages of both homogeneous and heterogeneous recycling modes have been indicated and investigated by different institutions. Most of the crucial issues are related to the fuel cycle characteristics and to the fuel forms. The different international studies often offer different perspectives and are based on different objectives, hypotheses and experimental results.

A recently initiated NEA study [7] aims at comparing the criteria for choosing between homogeneous and heterogeneous recycling modes as well as specific scenarios for implementation, potential non-proliferation issues and strategies for curium management. Moreover, the study will indicate the potential impact, both on the reactor core and on the power plant. Fuel and target related issues will be summarized with respect to potential limitations on, for example, the maximum allowable minor actinide content, residence time, helium production and remote fabrication implications. This evaluation will reflect previous, as well as ongoing or planned, irradiation programmes. Specific scenario studies will also be suggested in order to underline specific needs and requirements, both for the short and long terms. The aim of the study is to reach a deeper understanding of, and consensus on, the key issues and potential limitations and to allow making recommendations for further analytical and/or experimental demonstrations if needed.

2.3.2. Partitioning and transmutation

An alternative to storing all spent nuclear fuel in deep geological sites is to employ partitioning and transmutation technology, using either critical or subcritical fast reactor systems, in order to reduce significantly the long term radioactivity and residual heat of nuclear waste. The development of the partitioning and transmutation technology has, for example, shown the potential of minor actinide separation (partitioning) by means of aqueous- and pyro-reprocessing, demonstrating high separation factors at the laboratory scale.

To provide a forum for discussing scientific and strategic partitioning and transmutation issues, the NEA has, since 1990, organized biennial information exchange meetings. The meetings cover the impact of partitioning and transmutation on waste management and geological disposal, technical progress in fuels and materials, related physics and experiments, system design, performance and safety, as well as fuel cycle strategies and transition scenarios.

The last NEA Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation was held in Japan in 2008 [8], where a total of 120 papers were presented from 14 countries and three international organizations. A special session on fuel cycle strategies and transition scenarios was also organized. The next meeting will be held from 1 to 5 November 2010 in San Francisco, United States of America, hosted by the Idaho National Laboratory. The Czech Republic has offered to host the 12th meeting in 2012.

2.4. Reactors

2.4.1. Sodium cooled fast reactor (SFR)

With respect to the safety performance of SFRs, one of the foremost objectives of the Generation IV International Forum is to design cores that can passively avoid damage when the control rods fail to scram in response to postulated accident initiators (e.g. inadvertent reactivity insertion or loss of coolant flow). The analysis of such unprotected transients depends primarily on the physical properties of the fuel and the reactivity feedback coefficients of the core.

The newly established NEA expert group will address the above-mentioned objective by performing a comparative analysis of the safety characteristics of two different core sizes: a large-sized core (3600 MW(th)) and a medium-sized core (1000–2500 MW(th)). For both cores sizes, three types of fuel are proposed: oxide, carbide and metal. This comparative study aims at indentifying the advantages and drawbacks of each concept based on nominal performances and global safety parameters, such as neutronic characterization of global parameters (k_{eff}, power and flux distributions, void effect, Doppler, etc.) and feedback coefficient extraction, as well as discussion and agreement on corresponding calculation methodology. The study is expected to be completed in 2011 and the final report will include recommendations to improve safety (i.e. to avoid severe accidents) and future work related to minor actinide management.

2.4.2. Heavy liquid metal technology

Lead and lead–bismuth eutectic alloys are chemically inert, have a high boiling temperature and have good natural circulation characteristics. However, in order to use them as coolants in advanced nuclear systems, some mechanical and chemical behaviour issues, such as corrosion and embrittlement, should be solved. To elucidate these and other issues, the NEA has published a comprehensive handbook on lead–bismuth eutectic alloys and lead properties, materials, compatibility, thermohydraulics and technologies [9].

The purpose of the handbook is to develop standards, identify areas where further studies are needed and help establish a common methodology for experiments and data analyses. Eight institutes and national laboratories from seven member countries contributed to the study. The handbook contains four chapters dedicated to heavy liquid metal properties, four chapters on materials and testing issues and two chapters summarizing key aspects of thermohydraulics, instrumentation and system technologies. The last three chapters present information on existing test facilities and safety guidelines and provide an interesting perspective on more open-ended issues.

In addition, the NEA has also launched a benchmark on thermohydraulic safety issues of lead alloy cooled advanced nuclear energy systems [10]. This benchmark focuses mainly on characterizing the thermohydraulic behaviour of such systems under steady state forced convection and under natural convection, which is of critical importance for system design development. The first phase, the steady state forced case, is in the final stage and the second phase, the natural convection case, was recently started. As a conclusion, it is expected that the underlying experimental data can be examined and qualified by using large scale facility modelling and simulations.

2.4.3. Advanced reactors with innovative fuels

Plutonium and minor actinide recycling in thermal and fast reactors is being studied in many countries with the aim of maintaining and developing fuel cycle options, which can be adjusted to changing demands and constraints. The challenge is to move towards an economically and socially sustainable nuclear energy system based on advanced reactors (e.g. advanced water cooled reactors, fast reactors and, perhaps, accelerator based, hybrid reactors) and new types of fuel cycle, which would help to minimize nuclear waste.

Within this context, the NEA has been organizing a series of international workshops on Advanced Reactors with Innovative Fuels to enhance information on related R&D activities and to identify areas and research tasks where international cooperation can be strengthened. The scope of the workshops comprises reactor physics, fuel material technology, thermohydraulics and core behaviour of advanced reactors with different types of fuel and fuel lattice. Reactor types considered are water cooled and fast reactors, as well as hybrid reactors with fast and thermal neutron spectra.

The first workshop was held in Switzerland in 1998, followed by one in the United Kingdom in 2001 and one in the USA in 2005. At the last workshop in Japan in 2008 [11], more than 70 participants from 11 countries participated. Particular goals of the workshop were to identify research and development needs and the roles which can be played by existing experimental facilities, as well as the possible need for new experimental facilities. The conclusions of the technical sessions were synthesized and discussed in a roundtable meeting on international cooperation to facilitate the introduction of new reactor systems.

2.4.4. Accelerator driven systems

Accelerator driven systems with a fast neutron spectrum are possible options for nuclear waste transmutation and they would allow managing minor actinides in a double strata fuel cycle strategy. For more than ten years, the NEA has been conducting various activities related to the development of accelerator driven systems. One of the activities is the organization of workshops on the Utilisation and Reliability of High Power Proton Accelerators, to present and discuss the most recent achievements in the areas of accelerator reliability, which is critical to building accelerator driven systems.

Following the fifth workshop on Utilisation and Reliability of High Power Proton Accelerators in 2007 in Belgium [12], the NEA decided to extend the scope of the workshop to cover not only accelerators but also subcritical system designs and component development. The title of the workshop was therefore changed to Technology and Components of Accelerator-driven Systems and the first such workshop [13] was held from 15 to 17 March 2010 in Karlsruhe, Germany, and hosted by KIT. More than 70 abstracts were submitted in the field of accelerator driven system accelerators, neutron sources, subcritical system development and related experiments.

3. STRATEGIC ISSUES

3.1. Trends in nuclear fuel cycle

The renewed interest in nuclear energy has provided the incentive to study new reactor types within, for example, the Generation IV International Forum. The introduction of generation III+ and IV reactors will require changes in fuel fabrication, reprocessing, waste conditioning, etc. To review current developments in the nuclear fuel cycle, the NEA has started action to update the earlier publication, Trends in the Nuclear Fuel Cycle: Economic, Environmental and Social Aspects, issued in 2001 [14].

The scope of the new activity will cover a comprehensive review of existing publications and the assessment of advancements in nuclear fuel cycle technologies, ongoing global international initiatives, and progress and programmes of individual countries, with in-depth analyses of specific case studies. While trends in the aforementioned features, from the past and present up to what will be required in the future, will be underlined from a technological perspective, particular emphasis will be placed on policy and strategy issues, and the reciprocal influence on the course of technological progress, as well as sustainability aspects associated with the nuclear fuel cycle, such as long term supply of fuel resources, radioactive waste management and potential links to nuclear proliferation. The updated report is expected to be published in the second half of 2010.

3.2. Fuel cycle transition scenarios

Transition from current thermal fuel cycles to long term sustainable fuel cycles is one of most important items on the agenda for 21st century nuclear R&D. To meet the demand from member countries, the NEA is conducting various projects related to fuel cycle transition scenarios.

A recently published state of the art report provides a framework for assessing national needs regarding fuel cycle transitions and outlines the timing of key technologies [15]. Future nuclear fuel cycles could effectively address radioactive waste issues with the implementation of, for example, partitioning and transmutation. Previous studies have defined the infrastructure requirements for several key technical approaches. While these studies have proven extremely valuable, several countries have also recognized the complex, dynamic nature of the infrastructure problem (e.g. severe new issues arising when attempting to transit from current open or partially closed fuel cycles to a final equilibrium or burndown mode). Recognizing that many of the transition scenario issues are country specific when addressed in detail, it is believed that there exists a series of generic issues related only to the current situation and to the desired end point. These issues are critical to implementing a coherent nuclear energy infrastructure.

The NEA has conducted a study of a regional scenario based on the European region [16]. This study considers the implementation of innovative fuel cycles associated with partitioning and transmutation in Europe. Regional strategies can provide a useful framework for implementing innovative nuclear fuel cycles. Appropriate sharing of efforts and facilities among different countries is necessary in today's context, as is taking into account proliferation concerns and resource optimization. The report shows that the expected benefits from partitioning and transmutation, notably the reduction of radiotoxicity and heat load in a shared repository, can bring advantages to all countries of the region concerned, even when different nuclear energy policies are pursued. The study also demonstrates that regional strategies tend to favour a nuclear 'renaissance' in some countries.

Another NEA study [17], devoted to the key strategic and policy issues in transition from thermal to fast neutron reactors, has been undertaken with the objectives of:

(a) Identifying opportunities and challenges associated with the implementation of transition scenarios in various contexts (e.g. growth or stagnation of installed nuclear capacity, small or large nuclear power plant fleet in operation and different domestic uranium and fuel cycle industry situations);

(b) Analysing policy and strategic aspects of transition scenarios;

(c) Drawing findings and conclusions for policymakers.

The study took advantage of the above-mentioned NEA state of the art. The report stresses that fast neutron systems, operated with closed fuel cycles, offer capabilities to enhance the security of energy supply through better use of the energy content of natural uranium and facilitates waste management and disposal through a reduction in the volumes and the radiotoxicity of spent nuclear fuel. Recycling uranium, plutonium and minor actinides in fast neutron reactors can multiply by 50 or more the energy extracted from each unit of natural uranium mined. Furthermore, it shortens the time during which radioactive waste requires stewardship.

However, the attractiveness of fast neutron systems and the relevance of transitioning from thermal to fast reactors vary from country to country. Key parameters affecting the cost–benefit analysis of transitioning include the size and age of the nuclear reactor fleet, the expected future reliance on nuclear energy, access to uranium resources, domestic nuclear infrastructure and technology development, and radioactive waste management policy in place.

The transition to systems based on fast neutron reactors and closed fuel cycles is a challenging endeavour. The management of fissile materials during the transition period requires careful long term planning to evaluate the dynamic evolution of mass flows in evolving systems and ensure continuing security of supply at all steps of the fuel cycle. In-depth analyses of requirements for materials and services are a prerequisite to embarking on transition scenarios and should be based upon reliable data and robust models.

Infrastructure adaptation is another key challenge. Building industrial capabilities adapted to the transition period might be difficult at the national level. Multinational facilities could provide opportunities for economies of scale and economic optimization, which would be impossible at the national level. International cooperation could also help in ensuring an adequate supply of fuel cycle services at the global level while limiting the risk of proliferation.

Governments, which are responsible for formulating energy policies, have a major role to play in facilitating the implementation of fast neutron reactors and closed fuel cycles when they are an integral part of their strategic choice. Adaptation of legal and regulatory frameworks, R&D programmes, education and training, and stability of global energy policy are key aspects of government involvement and responsibilities.

3.3. Research needs for SFRs

In 2008, the NEA launched an activity to identify and prioritize research needs for advanced reactors, especially for gas cooled reactors and SFRs. The objective is to highlight facilities for safety research related to advanced reactors, as these reactors incorporate design features, materials and safety provisions that are likely to require exploratory experiments, verifications and confirmatory tests. Suitable facilities and an adequate infrastructure of expertise will be required to support safety evaluation and licensing for these reactors. This activity uses phenomenon identification and ranking tables and questionnaire findings to propose a strategy for the efficient utilization of facilities and resources to meet safety research priorities.

4. CONCLUSION

The nuclear energy renaissance expected in the first decades of the 21st century is likely to reinforce the attractiveness of fast neutron systems. Ambitious R&D programmes have been undertaken at the national level in many countries and in the framework of several international projects; they should lead to the design and development of advanced reactors and fuel cycle facilities responding to the sustainable development goals of governments and society. The OECD/NEA will continue to support member countries in the field of fast reactor development and related advanced fuel cycles by providing a forum for exchange of information and various other collaborative activities.

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ADVANCED CONCEPTS AND COOLANT TECHNOLOGIES

(Plenary Session 3)

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ADVANCED AND INNOVATIVE REACTOR CONCEPT DESIGNS, ASSOCIATED OBJECTIVES AND DRIVING FORCES

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Abstract

Advanced and innovative options for fast reactors are presented through a short selection of recent publications at international conferences. Driving forces and major trends are analysed to give a comprehensive overview of the various existing projects and supportive R&D.

1. INTRODUCTION

Nuclear energy appears more and more as an option which cannot be ignored in the quest for solutions that meet the increasing world energy demand, while reducing the release of greenhouse gases. Today's global installed nuclear capacity amounts to some 370 GW(e), which represents about 15% of the world's electricity generation. In the coming decades, nuclear electricity production will mainly originate from third generation light water reactors (LWRs), which are safe, reliable and efficient.

To date, LWRs consume less than 1% of natural uranium and the issue of uranium resources will become more acute as the size of the LWR fleet grows bigger. As for the spent fuel, some countries have recourse to an open cycle, while others have adopted a closed cycle with spent fuel treatment and partial recycling. The open cycle leads to storage, then to geological disposal of the bulk of the spent fuel, which is an option that does not seem compatible with a strong increase of the nuclear fleet in a large number of countries. The closed cycle allows for the sorting out of the different components of the spent fuel: uranium and plutonium can be recycled once in LWRs, which enables a 20–30% saving of natural uranium consumption, the final wastes being conditioned within glass and stored under simpler and safer conditions. However, LWRs do not allow for plutonium multi-recycling; used MOX fuel can be stored for further recycling, in fast spectrum rectors, for the plutonium it contains.

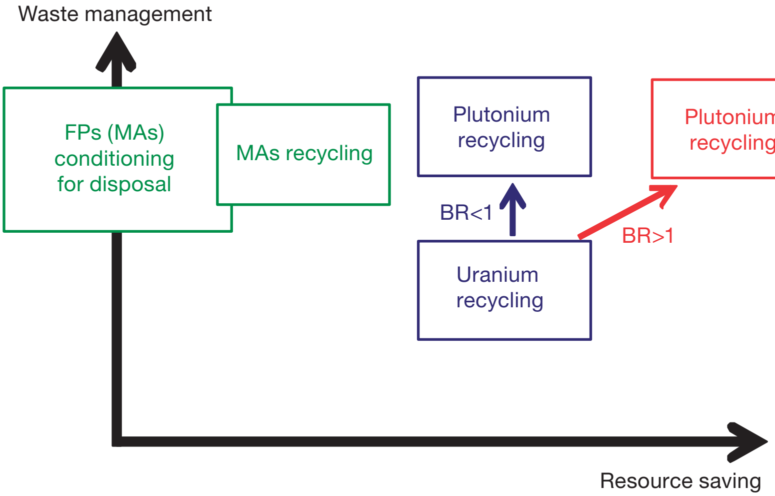

FIG. 1. Respective contributions to resource saving and waste management for every consitutent separated and/or recycled. FP: fission products, MA: minor actinides.

Fast neutron reactors (FNRs) enable the expansion of nuclear energy while meeting sustainable development criteria: resource saving and more complete waste management. Figure 1 shows the respective contribution to resource saving and waste management of every constituent separated and/or recycled.

Many countries are interested in this promising reactor type (sodium cooled FNRs) for which extensive feedback experience exists, corresponding to tens of reactors worldwide. This is the most mature technology. Heavy metal cooled FNRs (lead or lead–bismuth eutectic) may be an alternative to sodium and gas cooled FNRs (helium) may offer the advantages of FNRs and for applications other than electricity generation, especially high temperature applications.

2. ADVANCED OPTIONS FOR SODIUM FAST REACTORS (SFRs)

It is quite impossible to review all the studies and innovations proposed by teams working on SFRs and associated cycles in the frame of a single paper, but some important innovative works and results that have been recently published are reported here to give an overview of the main trends of research in these fields.

2.1. Fuel for SFRs

Fuel is a major issue for the development of SFRs, as it is strongly connected with reprocessing, recycled fuel fabrication, core design and safety.

Much experience has been gained with oxide fuel in the MOX form (plutonium and uranium oxides) and R&D is performed on addition of minor actinides to oxide fuels. Several solutions are envisioned in France to recycle minor actinides in oxide cores [1, 2]. Minor-actinide-bearing blankets are designed for heterogeneous recycling, a totally decoupled fuel/actinides management, by putting minor actinides in radial blankets on a depleted UO_2 matrix. Minor-actinide-bearing blankets have to be qualified to withstand cold temperatures and helium production. Minor-actinide-bearing fuels are designed for homogeneous recycling with low amounts of minor actinides (1–2%): the behaviour of fuel is quite similar to that of MOX, but all the fuel has to be fabricated by remote handling. An intermediate way is to have minor-actinide-bearing fuels with higher minor actinide levels located only in the outer part of the core. This allows for a significant gain in neutron flux while concentrating minor actinides in zones where sodium void contribution is of less importance.

Most the metal fuel experience has been gained with the EBR II [3]. R&D is also carried out on this fuel in the United States of America [4], Japan [5] and other countries. The US results show the technical feasibility of metallic fuel for minor actinide management. The transmutation fuels appear, on the basis of available results, to behave similarly to the ternary fuels (UPuZr), which have been used extensively. In Japan, oxide fuel is being considered for the short and mid-terms, while metal fuel is considered for future commercial reactors. Japan is considering some major innovations, such as low Zr metal fuel, lined cladding and He-bond particulate metal fuel. Low Zr metal fuel (Zr < 10%) allows for an increase in the fissile density and for an easing of the Zr recovery in the electro-refining process. Lined cladding may eliminate fuel cladding chemical interaction by putting a barrier inside the cladding tube. He-bond particulate metal fuel is very innovative; the fabrication process does not need casting moulds and the particles need to be sintered by low power preconditioning before full power reactor operation.

Less extensive experience has been gained with carbide fuel than with MOX and metal fuels, but it is considered because it has both high conductivity, as has metal fuel, and high fusion temperature, as has oxide fuel. India has gained experience on carbide fuel with the FBTR [6]; ten fuel subassemblies (70% PuC + 30% UC) have reached 155 GW·d/t without cladding failure. Several carbide core configurations have been studied in France [7]; one of them, with improved performances, has an initial plutonium mass of 5.1 t/GW(e) (–40% compared with the initial mass of a similar oxide core) for a core power density of

365 W/cm^3. The study of the unprotected transients shows an overall behaviour comparable to that of the oxide core but with a significant increase in margins before sodium boiling and before fuel fusion or dissociation.

Russian teams [8] also mention nitride fuel, which has high density and conductivity and allows for good performance and safety features. However, their analysis is that the helium bond does not allow for high burnup, while the sodium bond does not allow for very low sodium void worth. Moreover, commercial adoption of this fuel would require extensive time and funding.

2.2. Core studies and safety for SFRs

In French studies on oxide cores [9], improvement of safety and breeding ratio has been obtained by using large pins and low amounts of sodium associated with reduced power density (~200 MW/m^3). Such cores reach breakeven without blankets and show improved safety behaviour when compared to last generation reactors; for instance, the sodium void worth is reduced to ~US $5 and the low reactivity loss is an advantage for reactivity accidents such as unprotected rod withdrawal. Russian teams [8] have reached the same conclusion and have achieved an additional reduction of sodium void worth thanks to a sodium plenum located above the core. Japanese teams reviewed the evaluation methodology development of Level 2 PSA for SFRs [10], available or under development; analytical tools are listed to cover all the phases of accident sequences. Applications of these tools to the JSFR are made using the modified fuel subassembly with inner duct structure and control rod guide tube to enhance molten fuel discharge from the disrupted core.

In the past, metal fuel cores were generally associated with small or medium power reactors, but, in order to enhance the economic potential, KAERI increased the rated power to 1200 MW(e) from the previous 600 MW(e) [11]. Parametric studies lead to the conceptual core with a conversion ratio of close to 1.0, low plutonium inventory (~5.5 t/GW(e)) and void reactivity worth about US $7.5.

Argonne National Laboratory studied a 1000 MW(th) burner reactor [12] using both metal and oxide cores and showed that a wide range of transuranic conversion ratios from ~0.2 (short cycles and high sodium void worth) to breakeven can be achieved. Integral reactivity parameters showed that metal cores have more favourable inherent safety features.

The superiority of metal or oxide fuel for safety issues remains an open question. On one hand, metal fuel has good inherent features during unprotected transients; on the other hand, ceramic fuel has a much higher fusion temperature.

2.3. Reprocessing and fuel fabrication

The closed cycle is inherently associated with fast neutron reactors. Both the CEA and AREVA consider [13] that solvent extraction has proven over decades, on a commercial plant scale, its performance, reliability and mastery of technological wastes. R&D work will allow for adapting, improving and completing theses technologies to cope with new recycling needs and new constraints. The COEX™ process, which does not separate pure plutonium, and the separation and recycling of americium, are two typical R&D topics in this field.

In the frame of the FaCT project, the JAEA proposes advanced aqueous reprocessing [14] based on the well-established aqueous reprocessing of LWR spent fuel and newly applied processes such as uranium crystallization and extraction chromatography for minor actinide recovery. The development of these technologies is to be completed by around 2015.

The Indira Gandhi Centre for Atomic Research [15] will use an aqueous reprocessing route for PFBR and other MOX reactors; among their objectives is the improvement in separating fission products and minor actinides as well as sol–gel based fuel fabrication methods associated with vibrocompaction, which are more amenable to remote operation. In order to increase the breeding ratio and reduce the doubling time, the metal fuel and pyrochemical processes will be implemented beyond 2020. A breeding ratio of 1.45 and a doubling time of nine years can be reached with a mechanically bonded binary U–15Pu alloy with a Zr liner between the fuel and the clad.

The USA [16] is changing its strategy from emphasis on near term deployment to long term, scientifically based R&D. A wide range of separation processes is being investigated with the aim of contributing to a variety of fuel management options for LWRs and future fast reactors. Consideration is being given, among others topics, to off-gas capture and immobilization (iodine, krypton and tritium) and aqueous processes for separation of long life fission products — americium and curium. Laboratory scale tests have demonstrated the reduction of irradiated LWR fuels for preconditioning before electrorefining. Advanced electrochemical processes continue to be a high priority and efforts to understand fundamental thermodynamic behaviour of actinides in molten salts have been initiated.

3. FAST REACTOR CONCEPTS WITH GAS OR HEAVY METAL COOLANT

3.1. Gas cooled fast reactor (GFR)

The GFR is presented by the CEA as a promising long term concept which combines the benefits of fast spectrum and high temperatures and use of helium as coolant [17]. Prefeasibility studies of a 2400 MW(th) (~1100 MW(e)) GFR were undertaken in 2007 using carbide fuel with ceramic cladding (pins or plates); breakeven core without blankets able to recycle uranium, plutonium and minor actinides; and an indirect conversion energy system using a binary He–N_2 or He–Ar mixture with a compact plate stamped heat exchanger. Safety is an important issue favoured by a low coolant void worth (<US $1) and decay heat removal is addressed by both active and passive devices. Optimization is in progress regarding operating temperature, severe accident analysis and possible use of a prestressed concrete pressure vessel. ALLEGRO, an experimental reactor in the range 50–100 MW(th) is proposed as a first realization of the GFR.

3.2. Lead cooled fast reactor

ELSY [18] is a European programme for a 600 MW(e) reactor with pure lead coolant and oxide fuel with U, Pu and minor actinides. The ELSY project has some very innovative design features, including: open fuel assemblies of square pitch fixed at their upper end in cold gas space, a handling machine that operates under full visibility, spiral tube steam generators located inside the main vessel with provisions against steam generator tube rupture and decay heat removal dip bayonet exchangers immersed in the cold collector and cooled by water or air with a helium layer between coolant and lead. Temperature is limited to 480°C and lead velocity restricted to 2 m/s to prevent corrosion.

3.3. Lead–bismuth fast reactor

The SVBR-100 is a 100 MW(e) reactor cooled by lead–bismuth and studied by IPPE in the Russian Federation [19]. The whole core is changed at each refuelling, which is conducted every seven years. At the beginning, the SVBR-100 uses enriched uranium oxide fuel (16%) and moves progressively towards closed fuel by reusing generated plutonium and minor actinides, the remaining ^{235}U and a decreasing additional amount of enriched uranium. The reprocessing of spent fuel with high fissile contents provides economic benefits related to the small amount of fuel needing to be reprocessed. The pyro-electrochemical methods of spent nuclear fuel reprocessing and vibrotechnology for refabrication of fresh fuel are envisioned.

4. PROLIFERATION RESISTANCE

During the next decades, expansion of nuclear energy will be provided by Generation III LWRs. When Generation IV fast reactors with improved features in economics, safety and sustainability become available, countries will be interested in employing such fast reactors.

Fast reactors have some specific features in the field of proliferation resistance when compared with LWRs: no need for uranium enrichment, inherent associated fuel cycle with plutonium and possibly minor actinide recycling and breeding blankets. The international community has to address these features with appropriate extrinsic and intrinsic safeguards, physical protection and guarantees of fuel services; the Global Nuclear Energy Partnership statement of principles provides some relevant orientations for this. The ultimate goal is to be ready to ensure that fast reactor deployment should be done satisfactorily with regard to security and proliferation resistance.

5. MAIN TRENDS AND DRIVING FORCES

It is clear that the SFR is the main stream for the development and mid-term deployment of fast reactors. The most mature fuel cycle concept is oxide fuel associated with aqueous reprocessing; this is based on experience gained with former or current fast reactors and industrial reprocessing as well as recycling for LWRs. This concept may be improved with increasing proliferation resistance and separation of minor actinides; carbide fuel may be a promising option but needs considerable work before industrial deployment. Metal fuel associated with pyroprocessing is also an attractive concept owing to high fissile density and capability for short cooled fuel reprocessing. Most of the existing experience has been gained in the USA on EBR II, but several other countries have significant R&D programmes on this topic; the issues are fabrication of slug with minor actinides, decreasing zirconium content, reduction of technological wastes and, more generally, industrialization of the processes.

Other coolants are envisioned over the longer term: helium for high temperature application potential and heavy liquid metals because of their chemical inertness compared with sodium. However, these alternative concepts need more significant development, including experimental reactors and prototypes.

The driving forces for fast reactors are, on the one hand, saving of resources and autonomy from uranium procurement, and on the other hand, waste management. Other driving forces will be economics, including investment and operating costs, as well as reliability.

For those countries that have important LWR fleets and that feel confident about uranium availability for several decades, while still questioning the economics of fast reactors, the burner concept devoted to waste management of a lasting LWR fleet may be an attractive option. For those countries that do not have an important LWR fleet for plutonium generation and that question uranium availability over the long term, the breeder concept with a high breeding ratio and a low doubling time appears necessary to permit a rapid expansion of nuclear energy based on fast reactor deployment to answer their growing needs.

Today, fast reactors are not only a topic for R&D but also a matter of concrete realizations; three countries are engaged in the construction of SFRs. China has built the CEFR, an experimental 65 MW(th) (20 MW(e)) reactor which is expected to reach criticality in the near future. India is constructing the PFBR, a 500 MW(e) prototype reactor, which should become critical in 2010, and has plans for four more units with improved economics and safety. The Russian Federation is constructing the BN800 (800 MW(e)), which is expected to reach criticality around 2014, and has plans for further commercial units.

France and Japan have plans to launch SFR prototypes in 2020–2030 and KAERI has a long term R&D plan aimed at constructing a demonstration SFR by 2028 [20].

The USA considers it premature to prepare near or mid-term realizations of prototypes and that more science based research is still needed.

6. CONCLUSIONS

A very important amount of R&D is performed all over the world, covering a large spectrum, from applied R&D in support of ongoing SFR construction to mid-term R&D in support of future sodium prototypes, as well as to more long term, science based R&D to support very innovative concepts using sodium coolant or other coolants — helium or heavy metals.

Accordingly, the envisioned planning differs from short term deployment to long term R&D. Moreover, some countries are focusing on resource saving and rapid deployment thanks to a high breeding ratio, while other countries are focusing on waste management in a burner configuration. In all cases, there is agreement that fast reactors are needed in the future to address growing energy needs in a sustainable, safe and secure manner.

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LIQUID METAL COOLANT TECHNOLOGY FOR FAST REACTORS

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Abstract

In the paper presented are results of comparative analysis and the choice of liquid metal coolants for fast reactors, the current status of studies on the physical chemistry and technology of sodium coolants for fast neutron reactors and heavy liquid metal coolants, namely, lead–bismuth and lead for fast reactors and accelerator driven systems. There are descriptions of devices designed for control of the impurities in sodium coolants and their removal as well as methods of heavy liquid metal coolant quality control, removal of impurities from heavy liquid metal coolants and the steel surface of components of nuclear power plants (NPPs) and relevant equipment. Attention is given to the issues of modelling of impurity mass transfer in liquid metal coolants and designing new liquid metal coolants for NPPs. Results of the analysis of NPP abnormal operating conditions are presented. The adopted design approaches assure reliable protection against accidents. Up to now, about 200 reactor-years of sodium cooled fast reactor operation and about 80 reactor-years of submarine reactor operation have been gained. The new goals for sodium and heavy liquid metal coolant technology have been formulated as applied to the new generation fast reactors.

1. INTRODUCTION

Studies on the liquid metal coolants of nuclear power plants (NPPs) as a special trend in NPP design were initiated in the 1950s at the IPPE (former USSR) under the leadership of A.I. Leypunsky. A wide range of liquid metals having adequate nuclear, thermal, physical and chemical properties and low vapour pressure at high temperatures were considered as coolant candidates. The possibility of creating NPP designs that eliminate the failure of vessels and coolant loss was studied.

At that time, requirements were formulated for the coolants of various NPPs, namely, sodium for fast reactors, and lead–bismuth for nuclear submarine reactors and spacecraft (sodium–potassium and lithium). The effects of coolant on reactor neutronics, technological, corrosion and thermohydraulic characteristics, as well as on toxicity and cost, were taken into account. These issues were considered in close connection with the general safety of NPPs (including issues of reactor physics, fire safety, technological safety and protection against toxicity) [1–5].

1.1. Alkali metals

Sodium was chosen as a coolant for NPPs with fast reactors in the former USSR and in all other countries where studies on fast reactors were carried out (France, Germany, Japan, United Kingdom, United States of America, etc.). This choice was made because of sodium thermal physics and the simplicity of technological procedures for maintaining specified coolant quality in the repair stage [5–7]. Sodium drawbacks (i.e. high chemical activity in oxidizing media and its intensive interaction with water, accompanied by the production of hydrogen gas, and induced radioactivity of ^{24}Na and ^{22}Na with half-lives of 15 h and 2.6 h, respectively) led to the introduction of a secondary (intermediate) circuit in the reactor heat removal system. This intermediate circuit prevents the penetration of water–sodium interaction products into the reactor core and the primary circuit exposure to high pressure in the case of water–steam circuit failure.

The properties of lithium are quite unique [8]. For instance, its density, which is the lowest compared with that of the other liquid metals, assures a low mass of coolant loaded into the spacecraft reactor. Its specific heat capacity, which is the highest compared with that of all other metals, allows for a decrease in coolant temperature rise in the core and an increase in the average temperature of the thermodynamic cycle and, hence, its efficiency. Besides, the lowest pressure of lithium vapour (compared with other alkali metals) makes it possible to decrease the mechanical load on the components and pipelines of NPPs at high temperatures. On the other hand, liquid lithium has significant drawbacks, namely a high melting point (~180°C) and corrosiveness caused by dissolved nitrogen. Lithium interacts with water and burns in air at high temperatures [9].

Use of sodium–potassium eutectic, owing to its low melting temperature (–12.3°C), allows for the simplification of NPP design and facilitates its operation [10]. Unfortunately, Na–K has poorer thermal physics properties than those of Na, higher vapour pressure of the eutectic components, high chemical activity causing its spontaneous ignition in air at moderate temperatures, and a

TABLE 1. CHARACTERISTICS OF LIQUID METALS CONSIDERED AS CANDIDATE COOLANTS FOR NPPs

Liquid metals	Abundance in the earth's crust (wt%)	Cost[a] (roubles/kg)	Chemical activity in the environment	Corrosiveness	Toxicity
Li	0.005	60–100	Lower than that of Na and K	Higher than that of Na and K	Higher than that of Na and K
Na	~2.4	1–3	High	Low	Low
K	~2.4	~4	High	Low	Low
Hg	$\sim 5 \times 10^{-4}$		Low	High	High
Pb	0.016	~1	Low	High	High
Bi	$\sim 10^{-5}$	40–50	Low	High	High
Eutectics:					
Na–K		3–5	High	Low	Low
Pb–Bi		25–30	Low	High	High

[a] Cost in roubles in 1980.

tendency to form potassium peroxidates in air as a result of potassium vapour condensation in the low temperature areas. Their interaction with potassium may cause explosions.

1.2. Heavy liquid metal coolants

Mercury was the first liquid metal used as a coolant in fast reactors (Clementine and BR-2). Because of its high toxicity, limited primary resources, high vapour pressure and adverse nuclear and thermal properties, mercury is no longer considered as a possible coolant for fast reactors. Nevertheless, in the 1960s, it was widely used as a modelling fluid in the experimental studies on heat transfer in the NPP components.

The attractiveness of lead–bismuth eutectic as a coolant is due to its moderate melting point (125°C), high boiling temperature (1638°C), eliminating the possibility of its boiling onset in the high temperature areas, and low chemical activity with respect to air, water and steam, thereby preventing explosions and fires. Low working pressure in the circuit increases the reliability and safety of the components, simplifies the design and manufacturing technology and significantly facilitates operation of the primary system components. In the stage of designing NPPs for nuclear submarines [11–14], the properties of lead–bismuth eutectic outweighed its drawbacks, such as corrosion and erosion activity with respect to structural materials, high density and viscosity, low heat capacity and conductivity, as well as polonium accumulation under radiation.

The interest of designers of the larger size NPPs with fast reactors using liquid lead is due to its low chemical activity with respect to water and air, its high boiling point (1745°C) and the insignificant rate of polonium accumulation under radiation. In addition to the drawbacks of lead–bismuth eutectic, the lead melting temperature is high (327°C), and its interaction with water may cause explosions.

Analysis has shown that a reactor core cooled by lead should be 'loose' because the lead flow cross-section should be higher by an order of magnitude than that for sodium. It means that liquid lead velocity is lower by an order of magnitude compared with that of sodium. It is notable that sodium was chosen as the coolant for a high power density, compact fast reactor core in the early stages of nuclear power, when the main goal was to achieve a high ^{238}U–^{239}Pu breeding ratio in fast reactors.

Solid experimental and analytical task oriented studies [1–26] led designers to consider the liquid metal system of the NPP as a complicated, heterogeneous, multicomponent system, and the technology of liquid metals (sodium and lead–bismuth) was developed taking into account issues of liquid metal interaction with other media and structural materials.

This paper presents the current status of studies on the physical chemistry and technology of sodium and heavy liquid metal coolants (lead–bismuth and lead).

2. PHYSICAL AND CHEMICAL PROPERTIES OF LIQUID METALS AND THEIR TECHNOLOGY FUNDAMENTALS

Studies carried out on the 'coolant–structural materials–cover gas' system included many stages, starting with determination of constants characterizing solubility of impurities in liquid metal coolants to formation of models of mass transfer in the liquid metal circuits, taking into account thermohydraulic modes. The results of studies are used as the basis for designing computer codes to forecast system behaviour in all operation modes of the NPP. This requires knowledge of both equilibrium and kinetic constants. Considerable information has been gained about the solubility of various impurities in liquid metals and their mutual influence on solubility, kinetics of reaction in the coolant, diffusion constants, liquid metal structure and the form of impurities. Yet, this information is insufficient for a complete description of the system. The basic factors influencing mass transfer in the liquid metal circuits have been determined [2, 4, 15–18]. One of these factors determining the steel corrosion rate is total solubility of its components in liquid metals. These generalized data for 316 stainless steel at 440–950°C and liquid lithium, sodium, lead, and sodium–potassium and lead–bismuth eutectics are presented [16].

On the basis of the results of analysis of the data given in Ref. [15], the following important conclusion can be made: the observed spread of data on the solubility of 316 stainless steel elements in a liquid metal coolant is most probably caused by colloid type oxide admixtures. These oxides increase the corrosiveness of liquid metals [17]. Hence, in order to decrease the steel corrosion rate in liquid metal coolants, technological impurities should be removed from liquid metals. This technology proved to be effective as applied to sodium and it has been successfully used in the operating reactors.

It should be noted that steel corrosion in sodium is affected not only by oxygen but also by carbon, hydrogen and even nitrogen cover gases, with the corrosion rate being increased in the case of the presence of both oxygen and hydrogen in sodium. Features of behaviour of the above impurities were studied and special systems and devices were designed to control impurities in the coolant and minimize corrosion of structural materials in the sodium [2, 7].

Corrosion of structural materials in heavy liquid metal coolants can be decreased by formation of the oxide protective coating on the steel surface. Significant efforts were made to choose structural materials and determine the conditions for their reliable operation. In order to assure these conditions, the technology for lead–bismuth coolant handling was developed, as well as methods and devices for its purification, quality control and maintenance.

However, diffusion of steel elements through the oxide coating is not the only way of steel corrosion product transfer to the coolant flow. It is required that the Sherwood number characterizing the mass transfer from the fuel element cladding subject to front corrosion is much lower than the Biot number evaluating the rate of mass transfer through the oxide coating. Analysis has shown that Fe_3O_4 does not greatly suppress corrosion of the fuel element cladding in heavy liquid metal coolants. Some other protective film is required in order to reduce significantly penetration of corrosion products into the coolant. Even in the sodium circuits of the BN-600 reactor, corrosion products are accumulated instead of constant operation of cold traps, although the amount of corrosion products penetrating sodium is much lower compared with the amount in the lead–bismuth system at the same temperature. Estimates have shown that in order to eliminate the possibility of formation of oversaturated solutions of corrosion products in the BN-600 sodium circuits, the cold trap capacity should be increased by several orders of magnitude.

As applied to facilities with a lead based coolant, methods of slag oxide reduction using a special gas mixture, including hydrogen, were developed. A detailed order of these procedures can be determined in the course of special studies, taking into account the design features of the facilities and their operating modes. Indeed, however perfect the coolant purification system is, suspensions will be formed in the non-isothermal liquid metal circuit under normal operating conditions of the NPP. It is entirely possible that the presence of some suspensions is necessary for decreasing the corrosion rate. It is clarification of all these aspects that forces the designers carrying out such studies to take into account the parameters of advanced NPPs.

Therefore, there are a variety of processes taking place in a coolant–structural materials–cover gas system. Under normal operating conditions, the coolant does not only carry impurities (both dissolved and suspended) along the circuit, but also significantly facilitates their interaction with structural materials and suspensions. Impurities enter coolants in the areas where their chemical potential in the coolant is lower than in structural materials and suspensions in contact with it (reactor core and high temperature downstream core, including inlet section of the intermediate heat exchanger (IHX)). In the circuit areas, where the chemical potential of the impurities in the coolant is higher than in structural materials and suspensions, impurities leave the coolant to enter structural materials and suspensions. In the components where coolant temperature decreases (IHX and steam generator (SG)) and critical oversaturation is reached, spontaneous formation of suspensions may occur, followed by their coagulation and precipitation on the steel surface. A detailed description of the processes depends on components of such a system and their shares, which are determined, in turn, by the ratio of intensities of the impurities' sources and sinks.

Under abnormal conditions (e.g. sodium leak into the atmosphere or water leak into the sodium in the SGs), the range of phenomena expands. The burning process is accompanied by the increase in the local temperature and formation of aerosols. In the case of water leakage into the sodium, the rate of steel corrosion in this area increases by several orders of magnitude. As a result, a small leak grows into a large one and a hot flame is formed that strongly affects the adjacent tubes of the SG, causing their failure [19, 20].

3. SODIUM COOLANT TECHNOLOGY

The most substantial impurities in sodium coolant are oxygen, hydrogen, carbon and their compounds, including products of sodium interaction with water, air and hydrocarbons (lubricant), products of steel corrosion taking place in the course of long term reactor operation (Fe, Cr, Ni, Mn and Mg) and radionuclides (including tritium) [2, 7]. Studies were undertaken on the sources of impurities, their intensity and the possible negative effects caused by the impurities in the course of NPP operation. On the basis of these data, justification of permissible content of impurities in coolant and cover gas was made and the OST State Standard for sodium was developed as applied to the stages of supply and NPP operation [21]. Industrial technology of large scale production and supply of sodium meeting these requirements was adopted.

Oxygen is the most hazardous impurity from the standpoint of corrosion of structural steel. The oxygen content in sodium supplied by the manufacturer should not exceed 50 ppm and in the stage of operation, taking into account possible oxygen penetration into the circuit during repairs and in the case of circuit integrity loss, its content is limited to a 10 ppm value. Apart from oxygen, corrosion of structural materials in sodium is caused by carbon, nitrogen and hydrogen.

Proceeding from a 10% permissible decrease in steel strength over the course of reactor operation, 20 ppm and 30 ppm standards of carbon content are recommended, respectively, for the primary and the secondary sodium. The carbon content in sodium supplied to the NPP from the manufacturer should not exceed 30 ppm.

The influence of hydrogen on steel corrosion is less than that of oxygen, although the combined effect of these impurities increases steel corrosion. On the basis of the approach used for oxygen, 0.5 ppm of permissible hydrogen content in sodium was adopted.

In the early stage of studies on sodium coolant, it was considered that nitrogen did not have any strong effect on the mechanical properties of steel. However, it was revealed later that steel nitride hardening occurred because of nitrogen present in the cover gas. In this view, the nitrogen content in the cover gas at a temperature of over 300° C was limited by a ~0.3 vol. % value.

In order to effect control of impurities in sodium and cover gas, sampling devices were designed with considerable attention paid to assurance of representativeness of samples and required sensitivity and accuracy of analysis. The following three designs were chosen for use from the large number of sampling devices under study: (i) tubular sampling device, (ii) distilling sampling device and (iii) semi-automatic device for radioactive sodium sampling. The minimum detectable contents are as follows: oxygen (oxide, hydroxide and carbonate forms) – 2 ppm, carbon (non-volatile forms) – 4 ppm, nitrogen (nitride forms) – 1.6 ppm and fluorides – 2 ppm.

Control of activity of radionuclides in the circuit showed that the representativeness and repeatability of analysis results increase with the use of tubular flow sampling devices installed in the primary circuit bypass line and an activity measurement technique without sodium melting.

Among the other methods of routine monitoring of impurities in sodium, the following approaches were mainly considered: plugging meter, diffusion membrane sensors and electrochemical methods.

Analytical studies showed that plugging meter readings depended on the following parameters: hole diameter, number of holes, coolant flow rate, rate of temperature decrease of coolant flowing through the holes, type of crystallized impurity, as well as plugging meter design. If measurements are made with a dissolved plug, then the fixed temperature of its dilution onset would be closer to the saturation point than the plugging onset temperature of the plugging meter in the cooling stage. Values of parameters assuring authenticity of plugging meter data were determined experimentally. Plugging meter calibration was made in terms of oxygen, hydrogen and sodium–water interaction products.

The method of control of thermodynamic activity of impurities in sodium using a diffusion membrane sensor is based on measurement of impurities flowing from the coolant through special membranes to the other media (vacuum, inert gas or special gas mixture) with controlled parameters.

For the purpose of control of the hydrogen content in sodium, nickel was chosen as the material for the diffusion membrane. It was determined that the flow rate of hydrogen from sodium flowing through nickel was directly proportional to the hydrogen content. Various methods were used for measuring the hydrogen flow rate. The best results were obtained using magnet discharge pumps. These pumps were used in the automatic hydrogen detectors (IVA-1). The main purpose, as applied to the industrial facilities, is to detect water leakage into sodium in the SG. This is the main system used for SG leak detection.

Control of carbon in sodium is made using sensors with the membrane made of armco iron. Sodium at 750° C is on the one side of the membrane and special gas is on the other side. This gas interacts with carbon on the membrane surface to form carbon oxide or methane. The measured amount of produced carbon oxide or methane is proportional to the thermodynamic activity of carbon in sodium. This system was used for studies on carbon behaviour in sodium.

Measurement of oxygen in sodium is carried out by electrochemical cells designed by the Central Institute for Nuclear Research (former German Democratic Republic) in cooperation with the IPPE (former USSR). Electrochemical cell characteristics were studied in experimental rigs. These cells were then installed in the primary circuit of the BR-10 reactor and in the secondary circuit of the BN-350 reactor. The error in oxygen content measurement by electrochemical cell is 20%.

Methods of on-line measurement of nuclide activity in the circuit and control of fuel element cladding integrity were developed. The methods for on-line control of nuclide activity in sodium coolant flow and the appropriate device (CeNa — caesium in sodium) were designed as applied for the BOR-60 reactor.

Activity measurements are made by a Ge–Li detector in the sorption volume of about 1 cm^3 located in the flow tube. Measurements carried out over 1–10 min give reliable data on specific radioactivity of caesium nuclides and, under certain conditions, on xenon as well.

For the purpose of control of the hydrogen content in the cover gas, techniques for impurity analysis based on standard equipment (such as LHM-8 gas chromatograph, Zircon, Baikal and VTI gas analyser) were designed and introduced on an industrial scale. These techniques are capable of monitoring the contents of oxygen, nitrogen and volatile carbon containing impurities within the 10^{-7}–10^{-3} volume fraction range. Control of hydrogen in argon required designing special gas sampling systems, including filters for entrapping sodium vapours, and gas blowers. In the BN-350 and BN-600 reactors, hydrogen content is measured by a hydrogen conductometric analyser (KAV-7).

Methods of analysis and appropriate devices were adopted in sodium supplying plants and in the BR-10, BOR-60, BN-350 and BN-600 reactors, where special chemical and radiochemical laboratories were set up.

Purification of sodium coolant by removal of impurities in the fast reactor heat removal system is carried out mainly by the cold traps. Studies were made on hydrodynamic features and the heat and mass transfer processes in cold traps. The typical characteristics of the cold trap operation include low coolant velocity, large flow cross-sections that decrease with impurity accumulation in the trap, significant temperature differences and an extended mass transfer surface. Impurity distribution in cooled flow sections is characterized by high non-uniformity. In the downward flow of cooled sodium, maximum local concentration of impurities in the sediment exceeds the average value by an order of magnitude or more. This flow mode was typical for the first cold trap models, and it caused their low impurity retention capacity. It was shown that the settling box, confining up to 25% of impurities, was an effective component of cold traps. If the final cooling section is located above the settling box embedded in the cold trap, then the coefficient of impurity confinement in the settling box increases up to 50%. The average mass concentration of impurities in the settling box reached 56% and the impurity volume fraction was 35%.

These results, as well as data from studies on heat and mass transfer in non-isothermal and isothermal filters and results of tests of various cold trap designs, have determined the national approach to cold trap design. In the cold trap, there should be three sections connected in series, namely: (i) a cooled settling box, (ii) a final cooling section and (iii) an isothermal filter. Cold trap tests showed the effective removal of oxygen and hydrogen from sodium (if the residence time of sodium in the cold trap exceeded 15 min, then the coefficient of impurity confinement was close to unity). The minimum content of oxygen and hydrogen in sodium after its purification using various cold trap designs was equal to the solubility of these impurities at 120–150°C at the cold trap outlet. At these temperatures, solubility of oxygen and hydrogen is equal to 3–5 ppm and 0.02–0.05 ppm, respectively. Removal of corrosion products, in particular carbon, from sodium is less effective.

A method of cold trap regeneration after accumulation of impurities (causing increase of cold trap pressure drop) was worked out. This method implies the conversion of high melting impurities (sodium oxide) confined in cold traps into a low melting caustic phase. The caustic phase formed in the course of regeneration flows down to the settling box of the trap. Since its density is high, its volume is much lower than that of the impurities in the trap before regeneration. The caustic phase can be removed from the cold trap to the special tank, if necessary. This regeneration method proved to be highly efficient and cost effective and was adopted in the BN-350 and BN-600 commercial reactors and recommended for use in the BN-800 reactor.

Studies on the distribution of radionuclides in the entire volume of standard cold traps of the BR-5 and BOR-60 reactors revealed variable cold trap capabilities for accumulating radionuclides. Cold trap effectiveness in removing various nuclides from sodium determined as the ratio of equilibrium volumetric activity values measured before and after purification is equal to 100 for ^{131}I, 7.1 for ^{65}Zn, 1.5 for ^{137}Cs and ^{134}Cs, 1.3 for ^{60}Co and ^{124}Sb and 1.0 (no removal) for all other nuclides. It was shown that entrapment of caesium isotopes in cold traps was mainly caused by their precipitation on carbon impurities accumulated in the cold trap.

The sorption purification method was proposed for effective caesium removal from sodium using graphite materials. Small-sized, expendable adsorption devices were put into the reactor core to replace the fuel subassemblies of the core or radial blankets [22, 23]. These were used many times in the BOR-60, BN-350 and BN-600 reactors for removal of caesium radionuclides in the course of reactor refuelling at a sodium temperature of 160–220°C.

Use of carbon–graphite sorbent–carbonizate, having a high coefficient of sorption of caesium radionuclides from sodium at temperatures of up to 300–320°C made it possible to design adsorption devices for coolant purification and circuit cleaning in either shutdown or operating reactors [24]. For the purpose of removal of caesium radionuclides and suspended impurities from the coolant, small-sized adsorption devices were designed and tested. These adsorption devices contained up to 600 g of activated carbonizate (650 kg/m^3 density, 870 m^2/g specific surface of pores and 2.5 mm average granule size) and a porous metallic filter of 10–20 μm fineness [25].

Original analytical and experimental techniques were used for modelling studies on the physicochemical conditions of impurities in coolant and cover gas depending on the system composition and its temperature under conditions of thermodynamic equilibrium and taking into account reaction kinetics.

Models of homogeneous and heterogeneous mass transfer of the impurities in sodium circuits were developed, as well as computer codes for analytical studies on integral mass transfer of hydrogen and tritium and products of corrosion of structural materials. It was demonstrated that tritium produced in the fast reactor mainly entered cold traps in the primary and secondary circuits [27, 28]. The tritium amount accumulated in the primary cold trap is 1.5 times larger than that in the secondary cold trap. The tritium amount released through the SG to the third circuit is about two orders of magnitude lower than that accumulated in the cold traps. This amount released to the atmosphere through the walls of the sodium circuits is two to three orders of magnitude lower than that accumulated in the cold trap. The estimated rate of tritium release from the BN-600 reactor operating at rated parameters is 33 Ci/a. The total tritium amount released through the third circuit with unrecoverable feedwater is 86 Ci/a. This tritium mainly enters the hydrosphere.

The mathematical model of the mass transfer of structural material corrosion products was designed as applied to non-isothermal sodium loops. This model takes into account all of the above processes and the polydispersal nature of suspension particles. It was shown that the maximum thickness of corrosion product deposits in the IHX from 20 years of continuous operation would be about 1.6 μm. The maximum rate of structural material corrosion (about 1 μm/a) is observed in the high temperature section.

The most serious problems are caused by water leakage into sodium in the SG, sodium leakage to the atmosphere and spontaneous burning of the impurities in the NPP. In order to minimize the consequences of water leakage into sodium, studies were made on the justification of the systems for early leak detection and their rapid termination. Special media were developed for extinguishing sodium fires and preventing burning sodium product release to the environment. Studies made on sodium leaks from fast reactor heat removal systems showed that there were no inherent sources of large defect formation. This was also confirmed by the experience gained over many years of operation. Nevertheless, the probability of large sodium leaks is taken into account in fast reactor designs and corresponding studies are carried out.

On 21 January 1987, abnormal operating conditions were detected by the system of measurement of physical and technological parameters in the BN-600 reactor, showing spontaneous deviation of these parameters from preset steady state values.

Starting from the physical nature of all events registered under abnormal operating conditions, the attachment of impurities accumulated on the gas system surface in the previous operation period was identified as the most probable cause of this event. Later, NPP managers took measures to eliminate the initial source of this mode.

The results of studies and gained experience were used for the development of safe techniques for carrying out various procedures, namely: cleaning of the components from sodium residues and their decontamination, sodium protection against oxidation in the case of opening of the circuits, the order of procedures on dismantlement and installation of the circuit components, waste disposal and utilization of spent coolant [8, 15, 20].

Various methods of coolant residue removal from the components and waste disposal were studied, namely: washing by steam/gas, alcohol, water/alcohol and water mist, water/vacuum and stripping under vacuum. Composition of reactants, conditions and modes of procedures (temperature, duration, etc.) assuring work safety are considered for each method.

Reprocessing of spent sodium coolant for its disposal is associated with a significant increase in the amount of end product, and therefore it is expedient to use this sodium in the new fast reactors or in another industry. Reuse of the primary sodium requires its complete purification to reduce the content of long lived radioactive products by four to five orders of magnitude. This sodium can be used as an intermediate product in industry, for instance, in lead tetraethyl production.

Development of designs of the new generation fast reactors with improved safety characteristics (BN-1200) requires carrying out additional experimental and analytical studies and R&D work in the areas of physical chemistry and technology of sodium coolants, including:

(a) Studies on physical and chemical processes and determination of fundamental characteristics showing behaviour of impurities in sodium circuits, taking into account chosen structural materials and NPP performance;
(b) Designing innovative instruments for monitoring the impurity content in sodium and advanced methods and devices for sodium purification, including design and optimization of sodium purification devices located in the reactor vessel;
(c) Comprehensive tests of SG automatic safety systems;
(d) Analysis of the processes in cover gas systems (formation and washout of deposits and mass transfer of impurities, including aerosols);
(e) Studies on tritium mass transfer and accumulation in NPP systems, development of methods of entrapment and reliable confinement of tritium produced by various technological procedures;
(f) Studies on the mechanisms of processes taking place in technological procedures after removal of the components from the primary circuit,

updating of technologies for cleaning sodium system components, including cold trap and sodium disposal and reprocessing;

(g) Scientific justification of sodium circuit technological modes when changing for higher coolant parameters.

4. TECHNOLOGY OF HEAVY LIQUID METAL COOLANTS (LEAD–BISMUTH, LEAD)

In early 1959, the 27/VT nuclear power plant (the terrestrial prototype of a submarine NPP) was put into trial operation at the IPPE and the first nuclear submarine of project 645 with a lead–bismuth coolant was commissioned in 1963. However, in 1968, at the beginning of the second core lifetime, an accident occurred in the submarine reactor plant (project 645) and some problems in the operation of the 27/VT plant were revealed. Analysis showed that the initial cause was a lack of knowledge about the coolant and its negative characteristics. There were neither systems for monitoring or control of coolant quality nor equipment for coolant and loop purification. During reactor operation, the impurities (mainly oxides of coolant components) accumulated in the circuit, causing an abrupt deterioration of heat removal in the core. In the first stage of mastering the lead–bismuth coolant, these factors caused a decrease in the NPP lifetime. Operation failures of the terrestrial prototype and the submarine reactor plant during the first stage, as well as studies made on experimental facilities, revealed the necessity for thorough research of the features of both coolant and coolant–structural materials in non-isothermal systems.

The lead–bismuth coolant is corrosive to structural materials; during plant operation it can be contaminated by solid impurities as a result of its interaction with structural materials and oxygen. Therefore, the following two main problems had to be solved to ensure long term fail-safe operation of the plant, namely: (i) maintaining the required purity of both coolant and the inner surfaces of the circuit components and (ii) ensuring the corrosion resistance of structural materials in contact with the coolant. To solve these problems, a large scale study programme was developed on lead and lead–bismuth eutectic characteristics, physicochemical processes in non-isothermal circuits, corrosion resistance of structural materials and sources of impurities and their influence on the plant's reliability.

In previous decades, heavy liquid metal coolants were studied in the Russian Federation as applied to the stationary fast reactors (SVBR, BREST, etc.) and accelerator driven systems. Apart from lead–bismuth eutectic, it is planned to use pure lead as a coolant for the BREST-OD-300 reactor. Lead is attractive because it is neither deficient nor expensive; its activity, caused by polonium, is about three orders of magnitude lower than that of lead–bismuth eutectic. However, the high boiling temperature (327° C) hinders its use as a coolant. Experimental studies confirmed the similarity of the main physicochemical processes in lead and lead–bismuth coolants. This fact allows using the experience gained in substantiating lead–bismuth coolant for studies on lead coolant [10, 12, 26].

Stationary nuclear power plants of the new generation differ significantly from transportable plants in scale, configuration, operating characteristics and lifetime. These factors determine new R&D areas which help to verify concepts, methods and efficiency of coolant technology.

The structure of the impurities and their amount in the coolant depend on the type of structural material, operating mode, plant design and its purpose. During plant operation, impurities are formed by diffusion of structural material components through protective oxide films, corrosion and erosion processes caused by coolant interaction with structural materials and formation of new elements in coolant irradiated by neutrons and protons. Impurities may enter the circuit from cover gas in cases of depressurization of the cover gas system, reactor refuelling, repair operations, SG leaks, etc. Experimental studies have shown that the intensity of diffusion of components of structural materials into the coolant through protective oxide films in operating the BREST-OD-300 reactor plant amounted to 50 kg/a. The impurities can either be dissolved in coolant or suspended as solid, finely dispersed particles. During operation, impurities consisting mainly of oxides of coolant components and structural materials interact with each other, with the coolant and with the structural materials. Therefore, mass transfer processes go on permanently in the non-isothermal loop.

The method of oxygen inhibition has been used to the ensure corrosion resistance of structural materials. It was decided, correctly, to use oxygen naturally present in the alloy as an inhibitor, and this determined further areas of study.

Since the oxide compounds of steel components (mainly Fe_3O_4) are the basis of protective films in this method, their stability is determined by oxygen thermodynamic activity, depending on the concentration of oxygen dissolved in lead–bismuth and lead.

During plant operation, the decrease in concentration of dissolved oxygen down to the value equal to or below its equilibrium with Fe_3O_4 is possible and this may result in destruction of the protective film. However, a concentration exceeding the equilibrium value is also undesirable since it may lead to accumulation of an inadmissible amount of coolant oxides. Therefore, it is necessary to control coolant quality during plant operation by maintaining a certain dissolved oxygen concentration.

On the basis of the results of analytical and experimental studies and experience gained in operating transportable NPPs, the main processes ensuring regular plant operation regarding the coolant were defined. These are: removal of lead oxide impurities from the coolant using hydrogen, coolant filtration to remove suspended impurities, oxygen inhibition of coolant for maintenance of the required level of oxidation potential, purification of cover gas from aerosols and control of coolant parameters.

Different methods and equipment have been designed and used for coolant quality control. These include sampling devices for coolant and cover gas and continuously operating analysers of impurities in cover gas. Sampling is fulfilled at regular time intervals and followed by analysis in the laboratory using various methods. To control the concentration of oxygen dissolved in the alloy, a sensor registering oxygen thermodynamic activity (activity meter) was designed at the IPPE on the principle of a galvanic concentration cell and a solid electrolyte. All experimental facilities and, later, industrial scale plants were equipped with such sensors, improving the reliability of studies. The use of the sensor made it possible to confirm the results of theoretical analysis aimed at determination of optimal thermodynamic conditions for operation of non-isothermal plants. In recent years, sensor improvement has been made which increases the reliability of readings.

Two groups of new methods of coolant purification can be specified. In the first group, there are methods allowing conversion of impurities and return of conversion products into the coolant. These methods are used for removal from the circuit of impurities based on coolant component oxides, mainly PbO and Bi_2O_3. Methods in the second group allow removal of solid non-recoverable impurities from the circuit and their collection in special devices that can be replaced. These devices may be kept in the circuit during the whole lifetime of the plant.

The most effective method in the first group is lower oxide reduction with hydrogen: $Me_xO_y + yH_2 \leftrightarrow xMe + yH_2O$. Efficient purification is achieved by injection of a hydrogen, water vapour and inert gas mixture into the coolant flow. Coolant flow brings gas bubbles containing hydrogen to the circuit sections, where impurities are accumulated and hydrogen reduces solid phase lead oxides to lead, which is returned to the coolant. This reaction condition excludes the possibility of reduction of the oxides, which form the basis of protective films on structural materials.

Special dispensers provide two component (liquid metal–gas) flow in the entire circuit. The dispensers form a finely dispersed gas mixture with gas bubbles of 10–100 μm diameter transferred by the coolant, even in the sections with a low downward velocity flow (~0.2–0.3 m/s).

Filtration is the most efficient method used in the second group. Two main processes are operative for filtration. These processes are mechanical capture of impurity particles from the coolant flow and adhesion entrainment of impurities in the whole body of filter material. Filters are designed for a definite amount of impurities. As impurities accumulate, filter effectiveness decreases without coolant flow termination. This feature of filters is very important, because it allows using hydrogen purification. Coolant flow brings gas mixture bubbles containing hydrogen into the filter; hydrogen reduces lead oxides to lead, thereby partially cleaning the filter. Finally, only impurities that cannot be reduced by hydrogen (iron oxides, particles of structural materials formed due to abrasion and welding and other impurities formed during long operation of the circuit) remain in the filter.

Thus, the concurrent use of hydrogen reduction and filtration ensures cleaning of the inner surfaces of the circuit from slag deposits, return of lead from slag to the coolant and confinement in the filter of slag residues that cannot be reduced by hydrogen.

Filtering cloth materials having low hydraulic resistance, high porosity, relatively high capacity for impurities and sufficient mechanical strength and heat resistance have been developed and tested. Cloth made of glass, metal or carbon fibres meet all the qualifying standards.

For normal operation of the reactor plants, it is necessary to maintain an optimal concentration of dissolved oxygen in the coolant. The feature of lead and lead–bismuth circuits is the decrease of dissolved oxygen concentration during operation due to the release to the coolant of components of structural materials (Cr, Fe, etc.) and elements generated under the effect of proton and neutron beams in accelerator driven systems. This is due to their higher affinity for oxygen than that of lead–bismuth eutectic. They interact with dissolved oxygen to form oxides. An increase or decrease in dissolved oxygen concentration intensifies mass transfer processes; this may cause blocking of the cold leg sections or dissociation of the protective films on structural materials in the hot sections. Systems and equipment were designed for control of dissolved oxygen concentration in the coolant by supplying gaseous oxygen or a mixture of water vapour and hydrogen into the circuit, and dissolving solid oxides of lead or bismuth.

When oxygen mixed with inert gas is supplied to the cover gas or directly to the coolant, only part of the oxygen is dissolved. The majority of oxygen interacts with coolant to form solid lead oxides deposited in various sections of the circuit. That is why it is recommended that this method be used only in the emergency case of an abrupt deoxidation of the coolant. Supplying a mixture of hydrogen and water vapour into the circuit and changing their partial pressure ratio P_{H_2O}/P_H can be used for either decreasing or increasing concentration of the oxygen dissolved in the coolant. Formation of solid phase oxides of coolant components is excluded. The most efficient is the method based on the use of solid phase lead oxides, initially subject to special technological treatment. By varying coolant temperature and/or the flow rate in the reaction vessel, it is possible to control the rate of dissolution of lead oxides.

At the present time, there is great interest in accelerator driven systems, for which lead and lead–bismuth eutectic are considered as targets and coolants. New elements generated by a proton beam in the coolant may take part in various chemical reactions affecting chemical equilibrium in the circuit. The most important are oxidation–reduction reactions that may result in the formation of insoluble oxides and reduction of Fe_3O_4, which is the basis of the protective film formed on structural materials. Elements having a lower affinity for oxygen than that of lead (Au, Ag, Pt, Hg, Os, Cu, T1 and Bi) would be dissolved without formation of oxides. Elements having an affinity for oxygen lower than that of lead but higher than that of iron (Re, As, Te, Sb, Co, Ni, Mo, Sn and Fe) would be dissolved or oxidized depending on their thermodynamic activities. Lanthanides, halogens, and alkaline elements (Be, Y, Sc, Al, Ge, Ti, Hf) having a higher affinity for oxygen than that of iron, may reduce Fe_3O_4 and destroy the protective film under certain conditions. In general, the analysis showed that for 10–20 MW targets, the influence of impurities on mass transfer may be considerable, and this requires a special study.

As a result of substantiation of the NPP with lead–bismuth and lead coolants, scientific grounds were developed for coolant handling, new structural materials were chosen or created and new methods and equipment designed to monitor and control coolant quality, as well as for the removal of impurities from coolant and circuit surfaces. Methods and equipment were tested thoroughly in many experimental facilities, including the terrestrial prototype of the full scale transportable NPP, and adopted in commercial plants. When using new technology, there were no failures caused by the coolant in the 705 and 705K submarine plants during their entire operational lifetimes (80 reactor-years).

The results of the integrated study of lead coolant technology allowed development of the draft Regulations on Lead Coolant Handling Technology in the BREST-OD-300 Reactor Plant. This document includes the principal measures on lead coolant handling technology taken in all stages of construction, startup and operation of the BREST-OD-300 reactor plant.

It should be taken into account that when changing from experimental facilities to the commercial NPP, the concentration of impurities increases several hundred-fold, while the intensity of their sinks increases by only a few multiples. Therefore, the coolant in the cold leg of the circuit becomes significantly oversaturated with the impurities and the solid phase formed in the coolant may reduce the flow cross-section of the core.

Experience gained in designing and operating the submarine NPP with lead–bismuth coolant shows that obtaining guaranteed availability of a reactor using heavy liquid metal coolant requires large scale R&D programmes aimed, primarily, at the justification of corrosion resistance of the protective coating of structural materials of the operating NPP and the development of liquid metal technology ensuring steel passivation.

5. ANALYSIS OF ABNORMAL CONDITIONS

Core damage in the project 645 nuclear submarine has been the most severe event to have occurred in an NPP with heavy liquid metal coolants. In Enrico Fermi, the BN-600 and PFR reactors' sodium temperature increase was detected in some subassemblies and some fuel element failures occurred in the BN-600. However, no failures of the fuel subassemblies were observed. In the cases of water and sodium leaks, the safe temperature mode of coolant and fuel was assured by reactor shutdown and decay heat removal systems. New technologies and cold trap design were used for coolant purification and component cleaning, i.e. removal of products of sodium interaction with air (in the primary circuit), water (in the secondary circuit) and impurities penetrating the BN-600 reactor core. Parameter values of the core were restored in 100 h of reactor operation at 85% power. Since then, the BN-600 reactor has been in operation for almost 15 years without any limitations on its power. The BN-350 and PFR reactors were in fault free operation after cleaning the sodium circuits up until their decommissioning.

The favourable behaviour of the reactor core under the above abnormal conditions resulted from the thermal properties of the liquid metal coolant and the timely response of designed safety devices to the abnormal processes on the basis that liquid metal technology assured a minimum rate of structural materials corrosion guaranteeing NPP design lifetime; specified thermohydraulic characteristics of the reactor under design operating conditions; identification of abnormal processes and elimination of their negative consequences.

6. DESIGNING LIQUID METAL COOLANTS

An extensive knowledge of the properties of liquid metals, a deep understanding of their microstructure and atomic dynamics, the physical and chemical processes in these metals and the experience gained in their handling allow us to consider the possibility of a positive correction of their properties on the basis of specified attributes using any given solvable additions, and this may seem promising from the standpoint of designing a new generation NPP [15].

Prevention of sodium fires is important. Sodium burning in the environment is caused by two factors, namely: (i) high pressure of sodium vapour at moderate temperatures and (ii) high oxidation heat combined with low gas thermal capacity. For this reason, sodium vapour burning in the air forms a radiation source above the liquid metal surface. This source causes intensive heating up and the evaporation of liquid metal supplies metallic reactant, thereby maintaining the burning process. Analysis of these factors shows that fires can be prevented by the dissolution in sodium of non-volatile components (such as lead), since the non-volatility of an impurity increases its fraction on the free surface of liquid metal in proportion to sodium evaporation and forms a diffusion barrier to sodium release to the surface and to the environment. Preliminary experiments on air interaction with sodium–lead, with a lead atomic fraction below 0.09, showed that the mixture produced was slightly oxidized in the air at 600°C.

In order to assure high quality passivation of steel in liquid lead, it is reasonable to study lead–potassium eutectic having a 9% potassium atomic fraction; its oxidizing potential upon saturation with oxygen being lower than that of lead deoxidized by iron. In this case, chromium oxide film can be used for protection of the fuel elements, since with the oxygen potential below that of Fe_3O_4, dissociation level Cr_2O_3 would be formed on the chromium steel surface. Its growth rate does not depend on the oxygen concentration in the medium and the coefficient of steel element diffusion in chromium oxide is 10^{-11}–10^{-14} cm^2/s at 500°C.

7. CONCLUSION

As a result of substantiation of an NPP with sodium and heavy liquid metal (lead–bismuth and lead) coolants, the scientific grounds and methods and equipment for coolant handling were developed.

Extensive knowledge was accumulated on the physical chemistry and technology of sodium coolant for fast reactors and a great amount of experience was gained in its use in the prototypes and commercial power units of NPPs, with sodium coolant used in both the primary and the secondary circuits: BR-5 (10), BOR-60, BN-350, BN-600 (former USSR), Rapsodie, Phenix and Superphenix (France), EBR II and FFTF (USA), and DFR and PFR (UK). In view of the construction of the BN-800 and the design of advanced NPPs with the BN-1200 reactor, it is necessary to continue the R&D work programme with the purpose of improving the safety, environmental ‘friendliness’ and cost effectiveness of these plants.

The technology of lead–bismuth eutectic was carefully developed in experimental studies on many test facilities, including a full scale terrestrial prototype of the transportable NPP and was adopted in a special NPP designed in the former USSR for projects 645 and 705 nuclear submarines that had no international analogues. In recent decades, new results of studies on lead–bismuth and lead technology have been obtained as applied to fast reactors (SVBR and BREST) and accelerator driven systems. As regards the new generation NPPs, factors such as scale, lifetime, layout and mode parameters determine the direction of research in liquid metal coolant technology. Their guaranteed availability with up-to-date NPP parameters requires a large scale R&D programme, including justification of corrosion resistance of protective coatings of structural materials in the course of NPP operation and development of liquid metal technology ensuring steel passivation.

Knowledge of the liquid metals' microstructure and understanding of the physical and chemical processes in these metals allows for the possibility of taking positive correction of specified attributes by using additives to modify these attributes.

Realization of the concept of nuclear power development requires designing highly effective industrial technologies with respect to liquid metal coolants for fast reactors, guaranteeing compatibility of chosen coolants with the structural materials of the reactor plant and carrying out thermohydraulic tests under standard and abnormal conditions.

Taking into account the achieved level of studies, one can state that, in spite of recent significant progress in studies on heavy liquid metals as alternative coolants for NPPs with fast reactors, the most effective development of nuclear power in the next 20 years can only be realized on the basis of sodium cooled fast reactors. The basic efforts should be aimed at the design and construction of an advanced, high parameter NPP with fast reactors.

Experience gained in developing and operating marine NPPs and the results of analytical and experimental studies obtained during the last 25 years make it possible to consider that heavy liquid metal coolants would play an important role in nuclear technology development in the 21st century, allowing improvement of economic parameters and safety characteristics and offering a solution to the problem of handling long lived components of radwaste using accelerator driven systems or fast reactors.

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SAFETY AND MATERIALS

(Plenary Session 4)

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DESIGN AND ASSESSMENT APPROACH ON ADVANCED SFR SAFETY WITH EMPHASIS ON THE CORE DISRUPTIVE ACCIDENT ISSUE

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Abstract

The safety of future sodium cooled fast reactors (SFRs) will be achieved at the same level as that achieved for future light water reactors (LWRs). The concept of defence in depth, as widely applied to the design of LWRs, will be applied to the safety design of advanced SFRs. Through the prevention, detection and control of accidents, core disruptive accidents (CDAs) will be excluded from design basis events. Considering that the SFR reactor core is not the most reactive configuration, unlike in LWRs, design measures to prevent CDAs and to mitigate the consequences of them are being considered as provisions for beyond design basis events. To meet future nuclear energy system safey goals effectively, advanced SFR designs should exploit passive safety features to increase safety margins and to enhance reliability, i.e. prevention and/or mitigation of CDAs. In particular, the safety approach needed to eliminate severe recriticality will be highly desirable, because with this approach, severe accidents in SFRs can be simply regarded as being similar to LWRs. In addition, it is easier to make full use of the excellent heat transport characteristics of sodium coolant in achieving in-vessel cooling and the retention of post-accident core debris.

1. INTRODUCTION

Many sodium cooled fast reactors (SFRs) have been designed, licensed and constructed in several countries so far. Operational experience on SFRs has been accumulating to total approximately 400 reactor-years. On the basis of the experiences gained, it could be mentioned that the SFR technology has matured sufficiently well to a level that such a reactor concept is licensable and deployable in any country. Although the SFR concept is one of the most promising candidates for next generation reactors, meeting the objective of sustainability, they need further investigation for making them economically competitive with the light water reactors (LWRs) from the same era. Therefore, it is necessary to keep in mind that achieving the required level of safety for next generation reactors has to be a rational process.

In future, the number of reactors in operation will have to be increased considerably compared with current numbers. Keeping the safety level of the newly deployed reactors at the same level as operating plants today would lead to an overall increase in the risk of nuclear accidents. The development of advanced SFRs will be directed at enhancing safety, aiming at meeting the safety requirement of next generation reactors such as the safety goal of the Generation IV International Forum [1] and the safety principles of the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO)[2].

This paper describes the key characteristics of SFRs, the historical approach to accommodate SFR specific safety issues, safety goals or principles for advanced SFRs, and the safety design and assessment approach on advanced SFRs to meet the safety requirements for next generation reactors.

2. KEY CHARACTERISTICS OF SFRs

The SFR is a fast neutron sodium cooled reactor that has inferior neutron absorption and moderation characteristics to conventional LWRs, which makes it possible to produce nuclear fuel or to burn Pu and minor actinides while generating electric power. Hence, the SFR is one of the most promising concepts for next generation nuclear systems.

The good heat transport characteristics of sodium can permit the design of a compact, high performance and low pressure reactor system. The primary coolant has a relatively large thermal inertia and a large margin with respect to coolant boiling is achieved by design. Decay heat removal (DHR) by natural circulation is possible only by ensuring the optimum layout of the cooling circuit with regard to the ultimate heat sink. This enables use of simpler and highly reliable DHR systems with no dependence on support systems. Another major safety feature is that the primary system operates near atmospheric pressure, pressurized only to the extent needed to circulate fluid. Therefore, it is easy to maintain coolant inventory for core cooling because there is no rapid flushing out of primary coolant, even upon primary boundary failure.

On the other hand, sodium reacts chemically with air and with water and thus the design must limit the potential for such reactions and their consequences. With regard to safety, a secondary sodium system acts as a buffer between the radioactive sodium in the primary system and the energy conversion system.

The core disruptive accident (CDA) is highly important in the commercialization of SFRs. Since sodium void reactivity becomes positive in a large reactor core, sodium boiling would result in a power increase under the anticipated transient without scram (ATWS). Furthermore, the fast reactor core is not in the highest reactivity configuration, recriticality due to the coherent movement of molten fuel core might lead to a significant release of mechanical energy. Thus, the recriticality issue in the CDA condition has been the most significant safety issue for the SFR from the beginning of its development history.

3. HISTORICAL PERSPECTIVE OF THE SAFETY APPROACH TO THE CDA ISSUE

A fairly coherent safety approach was taken in the SFRs developed in the 1970s and 1980s; Super Phenix [3, 4], SNR-300 [5], CRBRP [6] and Monju [7]. These plants were designed on the basis of defence-in-depth (DiD) principles with appropriate consideration of SFR characteristics. The conventional safety approach to the CDA issue is: (i) to minimize the probability of occurrence of CDAs by utilizing, for example, two independent reliable reactor shutdown systems, and (ii) to assess the mechanical energy release due to recriticality events, assuming a hypothetical CDA, confirming the integrity of the reactor vessel and component against the estimated mechanical energy and/or loading due to burning of sodium that could be spilled out from the reactor vessel.

The SFRs designed during the 1990s have incorporated many innovative design ideas, e.g. EFR, BN-800, Advanced Liquid Metal Reactor and DFBR. The so-called ‘risk minimization’ approach taken in the EFR [8] incorporates a ‘third shutdown level’ which is capable of maintaining core integrity in case of postulated failure of two basic shutdown systems. Through these preventive measures, the risk of core melt is extremely low. Nevertheless, according to the ‘as low as reasonably achievable’ principle, both primary and secondary containments are installed, which further mitigate against postulated loading.

The BN-800 [8] reactor design is based on the BN-600, but is an improvement that uses the experience gained of BN-600 reactor operation and accommodates enhanced safety features. The safety design changes and modifications compared with the BN-600 are an additional passive shutdown system with hydraulically suspended rods, a special sodium cavity over the core to reduce the sodium void reactivity effect and a core catcher for collecting core debris in the case of its melting. In a licensing procedure, it is reported that CDAs were evaluated, given their occurrences, to some extent for various accident initiators as a part of beyond design basis event (BDBE) safety assessment.

The US Advanced Liquid Metal Reactor [9] is a small modular reactor with a metal fuelled core which has passive safety features, namely, core radial expansion and axial expansions of control rod drive lines and the reactor vessel design. The gas expansion modules (GEMs) are added to mitigate uprotected loss of flow events. In addition, the active shutdown system has a diverse system known as the ultimate shutdown system, in which B_4C absorber spheres are dropped manually into the central channel of the core. Despite the many active and passive preventive measures, representative CDAs are still evaluated to demonstrate that occurrence of core disruption is unlikely and to determine the margin of the containment.

Japan's DFBR [10] incorporates a passive reactor shutdown feature known as the self-actuated shutdown system (SASS), where a Curie point magnet is adopted in the magnetic circuit to de-latch absorber rods. The treatment of CDAs was taken from Monju, namely, they are regarded as a BDBE category and evaluated on a best estimate basis to confirm a safety margin of the plant.

Even though none of these plants, with the exception of the BN-800 in the Russian Federation, were actually licensed or constructed, many of the advanced and innovative design concepts developed are extremely useful as the technological basis for designing future SFRs.

Enormous efforts have been dedicated to the clarification of the accident scenario and the consequences of CDAs, especially the uprotected loss of flow and unprotected transient overpower scenarios that have been historically investigated from the viewpoint of mechanical design margin against the power burst during the initiating phase and energetic recriticality during the transition phase. In parallel, the safety assessment method of the CDA has been improved from the very beginning with the Bethe-Tait model in 1956, which assumes the gravitational fall down of the core fuel, to the recent more mechanistic models such as the SAS 4A and the SIMMER-III code [11]. The mechanistic models consider various material motion and phase change mechanisms based on various in-pile (e.g. TREAT, CABRI) and out-of-pile experiments. Over time, thanks to severe accident R&D progress, the mechanistic analytical approach has been improved with the evolution of safety knowledge and has reduced the mechanistic energy release as shown in Fig. 1. Even though their early designs consider CDAs directly in the safety design, CDAs are treated in the safety evaluation as BDBEs, with best estimate methods and assumptions.

4. SAFETY DESIGN GOALS/PRINCIPLES FOR ADVANCED REACTORS

The current safety requirements are described in IAEA Safety Standards Series No. NS-R-1, which takes account of the developments in safety requirements by consideration of severe accidents in the design [12]. A new safety risk informed approach was proposed for new reactor design [13]. The safety of future SFRs will be achieved at the same level as that achieved in future LWRs. The safety design for advanced SFRs will be taken into account in the evolution of

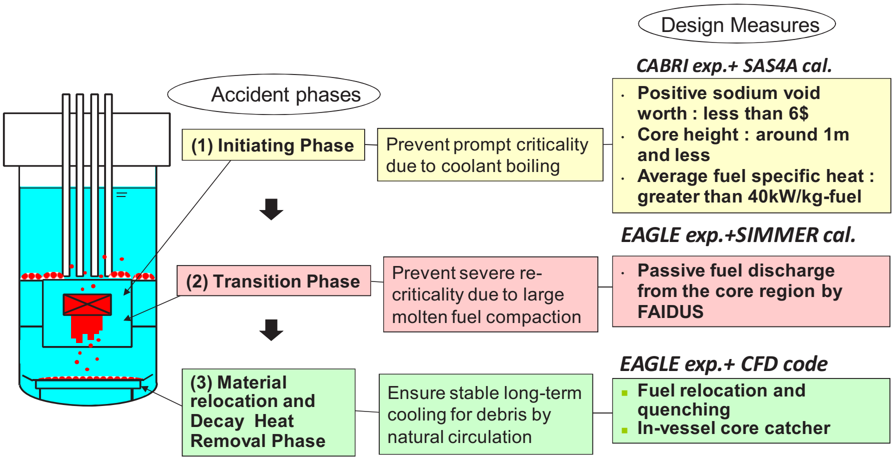

FIG. 1. Historical transition of evaluated CDA mechanical energy release.

safety requirements, with reference to the safety design goal/principle defined in international forums such as Generation IV and INPRO.

4.1. Generation IV

Generation IV nuclear energy systems are being developed under the initiative of the Generation IV International Forum begun in 2000. The SFR was selected as one of six promising concepts. Three goals for the Generation IV nuclear systems have been defined in terms of safety and reliability, as listed below [14]:

(1) ***Safety and Reliability 1, Generation IV nuclear energy systems' operations will excel in safety and reliability.*** The focus of this goal applies to safety and reliability during normal operation of all facilities employed in the nuclear fuel cycle and, thus, deals with the relatively likely kinds of operational events that set the forced outage rate, determine worker safety and result in routine emissions that could affect workers or the public.

(2) ***Safety and Reliability 2, Generation IV nuclear energy systems will have a very low likelihood and degree of reactor core damage.*** This goal calls for design features that create high confidence that the possibility of core damage accidents will be very small for Generation IV reactors. The goal deals with minimizing the frequency of initiating events and with provision of design features that ensure that the plants can successfully control and mitigate any initiating events that might occur without causing core damage.

(3) ***Safety and Reliability 3, Generation IV nuclear energy systems will eliminate the need for off-site emergency response.*** It is desirable that Generation IV systems demonstrate, with high confidence, the capability of the safety architecture to manage and mitigate the consequences of severe plant conditions and that any potential releases of radiation will be small and have only insignificant public health consequences.

4.2. INPRO

INPRO was initiated in 2000 under the auspices of the IAEA. The basic principles for the safety of innovative nuclear energy systems (INS) have been established in INPRO, taking into account the large body of work that already exists on dealing with the safety of reactors and fuel cycle facilities currently operating and the previous work carried out on establishing requirements for next generation reactors [15].

- **Safety Basic Principle BP1**: Installations of an INS will incorporate enhanced DiD as a part of their fundamental safety approach and ensure that the levels of protection in DiD will be more independent from each other than in existing installations.
- **Safety Basic Principle BP2**: Installations of an INS will excel in safety and reliability by incorporating into their designs, when appropriate, increased emphasis on inherently safe characteristics and passive systems as a part of their fundamental safety approach.
- **Safety Basic Principle BP3**: Installations of an INS will ensure that the risk from radiation exposure to workers, the public and the environment during construction/commissioning, operation and decommissioning are comparable to the risk from other industrial facilities used for similar purposes.
- **Safety Basic Principle BP4**: The development of INS will include associated research, development and demonstration work to bring the knowledge of plant characteristics and the capability of analytical methods used for design and safety assessment to at least the same confidence level as for existing plants.

5. SAFETY DESIGN AND ASSESSMENT APPROACHES FOR ADVANCED SFRs

Different levels of logical defence lines are considered in the DiD concept. Employing a definition of the IAEA, this concept for nuclear power plants consists of the following five levels [15]:

- Level-1: Prevention of abnormal operation and failures.
- Level-2: Control of abnormal operation and detection of failures.
- Level-3: Control of accidents within the design basis.
- Level-4: Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents.
- Level-5: Mitigation of the radiological consequences of significant releases of radioactive material.

A basic safety approach in designing advanced SFRs is essentially the same as one taken in LWRs. The concept of DiD, as widely applied to the design of LWRs, will be applied to the safety design of advanced SFRs.

5.1. Design basis

The three first levels of the DiD are prevention, detection and control of accidents, which are termed ‘within design basis’. Accident prevention is the first priority, because provisions to prevent deviations of the plant state from well-known operating conditions are generally more effective and more predictable than measures aimed at mitigation of the abnormal conditions. With a primary emphasis on preventing and detecting abnormal occurrences, safety design provisions will be provided for control of postulated abnormal conditions, which are the appropriate means to shut down the reactor, cool the residual heat in the reactor core and contain radioactive materials within the reactor facility. CDAs will be excluded from design basis events (DBEs) by the safety provisions for the first three levels of DiD. Ensuring the independence of different levels of protection is a key element in avoiding the propagation of failure into subsequent levels. This might be accomplished by more extensive use of inherent safety characteristics, by more use of passive systems and/or by more use of diverse systems.

After the design phase, detailed analysis and assessment of the safety architecture are required to ensure that all challenges and mechanisms are correctly addressed and that in any DBEs, sufficient barriers remain effective to meet the radiological objectives with due reliability. The comprehensive identification of initiating events and the following analysis to assess their potential consequences allow identification of the set of representative postulated initiating events (PIEs). The safety assessment in DBE will be treated in a conservative manner.

Even though the philosophy involved in the DiD concept has been universally accepted, the SFR specific issues will be taken into account in the technical implementation. PIEs should include sodium fire, sodium–water reaction, local fault, etc., taking into account the safety characteristics of the SFR.

5.2. Beyond design basis

For the purpose of meeting the third design goal (safety and reliability 3) of the Generation IV International Forum, eliminating the need for the fifth level of DiD, there is a need to strengthen the safety design of the fourth level of DiD, which is severe accident management. It should be noted that this design goal would not exclude the preparation of an emergency response plan. Actually, there is the fact that emergency response plans have already been prepared in compliance with national laws and regulations in many countries. In this sense, it is essential to provide design measures to: (i) prevent accident progression and (ii) mitigate postulated severe accidents within a plant and/or to at least provide a sufficient period of grace before core damage occurs and/or containment failure for the recovery by the operator and for the emergency response procedure decided upon by the authority, taking into account the characteristics of severe accident progression.

In level 4 of DiD, namely, as a BDBE, typical event sequences such as ATWS and loss of the heat removal system could be considered. Because the ATWS event sequences progress in the order of minutes, it is difficult to depend on any operator's actions to terminate the accident prior to core damage. Since sodium void reactivity tends to become positive in a large reactor core, sodium boiling would increase power in the event of progression of ATWS. It is important to provide the provisions primarily to prevent sodium boiling or CDA given ATWS conditions such as the failure of two reliable reactor shutdown systems.

The fast reactor core is not in the highest reactivity configuration, recriticality due to the coherent movement of the molten fuel core might lead to a significant power burst resulting in a high level release of mechanical energy. If a recriticality event with such a release is considered, the containment of such a sequence will be economically unfeasible and publicly unfavourable. Therefore, it is desirable to eliminate the recriticality following such a release by design.

The loss of heat removal system event sequences progress rather gradually, and a sufficiently long period to core damage allows the operator to undertake recovery action and/or accident management. Making maximum use of passive safety features such as natural circulation capability to remove decay heat and making some provisions for accident management, such as alternative systems/components, is the rational safety approach and could allow these sequences to be considered a residual risk.

Because the sequences categorized in BDBE are very low probability events, acceptable provisions need not involve the application of conservative engineering practices used in setting and evaluating DBEs, but rather should be based on realistic or best estimate assumptions, methods and analytical criteria.

The purpose of CDA analysis has, therefore, been to provide or confirm an additional safety margin of the plant and the effectiveness of provisions for control of CDAs.

5.3. Key safety issues for advanced SFR safety

5.3.1. Passive safety

To meet the future nuclear systems' goal effectively, advanced SFR designs should exploit passive safety features to increase safety margins and to enhance reliability. A number of design ideas preventing core damage have been proposed and investigated so far. The passive safety features considered are the reactor core with inherent negative reactivity feedback effects, the passive reactor shutdown systems and the DHR system via natural circulation. In addition, mitigation of CDAs can be strengthened by provision of passive safety features.

The reactor core concepts aimed at prevention of core damage are proposed taking into account reactivity feedback due to axial fuel expansion and radial core expansion, control rod driveline expansion or a special sodium cavity over core to reduce the sodium void reactivity effect. Given the ATWS condition, coolant temperature rise would result in reactor power reduction due to negative reactivity feedback. There is an extensive technology base in nuclear safety that establishes the passive safety characteristics of the SFR and their capability to accommodate all of the classical ATWS events without fuel damage. Landmark tests of these events were done in RAPSODIE (France) in 1983 and in EBR-II, and FFTF (USA) in 1986. However, the system behaviour will vary depending on system size, design features and fuel type and thus their function and effectiveness should be demonstrated and their reliability should be confirmed.

The possible design features to enhance reactor shutdown function are the self-actuated shutdown system (SASS) with Curie point magnet using the temperature sensing alloy, the passive shutdown system with hydraulically suspended rods and the gas expansion module (GEM). Given the ATWS condition when the core outlet coolant temperature rises, the sensing alloy temperature reaches the Curie point. Absorber rods are de-latched owing to the decrease of magnetic force and are inserted into the core. The basic characteristics of SASS have already been investigated by various out-of-pile tests for material elements. As the final stage of the development, the stability of SASS has been confirmed under the actual reactor operational environment with high temperature, high neutron flux and flowing sodium to ensure the high plant availability factor [16]. Hydraulically suspended rods that are suspended by coolant flow during normal power operation would drop into the core by the decrease in coolant flow due to pump coastdown. The GEM is installed in the negative void reactivity region of the reactor core. If a primary pump coastdown occurs due to some cause without reactor scram, the pressure would drop due to loss of pump head and gas volume would expand inside the GEM, resulting in insertion of negative reactivity.

The most significant design feature is a passive DHR system via natural circulation. The reliability of DHR could be considerably improved by utilizing passive safety features. A relatively small number of components, which are mostly static, will perform their mission because they do not require the support systems, such as an electrical power supply and component cooling system. This feature would be utilized not only to effect control within a DBE but also to control a BDBE in the advanced SFR. Post-accident DHR could be achieved for a degraded core by natural circulation. It might be difficult to keep active components available during the CDA progression. It is easier to make full use of the excellent heat transport characteristics of sodium coolant in achieving in-vessel cooling and retention of post-accident core debris.

5.3.2. Mitigation of CDA consequences

The favourable passive safety behaviour of SFRs is expected to exclude almost completely the possibility of severe accidents with the potential for core damage. Considering the SFR reactor core is not the most reactive configuration, unlike in LWRs, nevertheless, design measures to mitigate the consequences of severe accidents are being considered according to the as low as reasonably achievable principle. This approach is consistent with the DiD philosophy of providing an additional safety margin against BDBEs.

Considering the commercialization era of SFRs when a number of large scale plants are deployed, it is strongly expected to not only confirm that the consequences of a CDA can be contained, but also to resolve such a major issue, especially a recriticality issue. For this purpose, elimination of a recriticality event in the course of CDA sequences has become one of the major goals in reactor safety R&D.

In particular, the safety approach to eliminate the severe recriticality will be highly useful, because with this approach, severe accidents in SFRs can be simply regarded as being similar to those in LWRs. In addition, it is much easier to maintain the integrity of the reactor boundary and coolant circuit necessary to

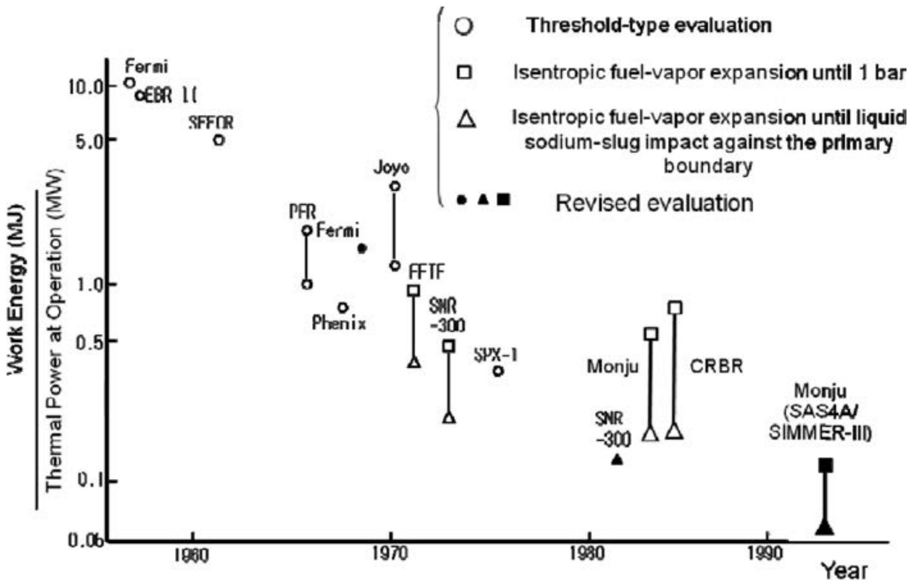

FIG. 2. Safety provisions for mitigation of CDAs.

remove decay heat. Also, no mechanical energy release means no challenge is posed by spilled sodium burning in the containment. Achieving this level of safety should result in licensing and regulatory simplifications that may, in turn, result in reduced system cost.

It is known through various CDA analyses of reactor cases that severe recriticality events can be avoided if some fraction of the initial core fuel inventory is discharged from the core region. This fraction depends on the fissile enrichment of the fuel. For a larger core with lower fissile enrichment, a lower fraction of the initial core fuel inventory is enough to prevent the recriticality event. In this approach, a special fuel subassembly design concept is adopted so that early fuel escape from the core in a CDA can avoid large scale molten pool formation, which leads to potentially severe recriticality [17].

In the JSFR design, elimination of severe recriticality in CDA sequences is addressed to control the potential of excessive void reactivity insertion in the initiating phase with appropriate design parameters such as maximum void reactivity and to prevent core-wide molten fuel pool formation by the introduction of a dedicated fuel subassembly with an inner duct (FAIDUS) as shown in Fig. 2. The effectiveness of these measures is being evaluated on the basis of in-pile tests such as EAGLE and various out-of-pile tests and computer simulations by validated analytical tools [18].

5.3.3. Probabilistic considerations

The deterministic safety approach is complemented by a probabilistic safety evaluation, which verifies design features that assure very high levels of public health and safety. A risk informed approach in the design stage is desired for attaining a well-balanced safety design. In the course of the design of the provisions, the consideration of reliability targets to cope with the probabilistic success criteria of each level of defence represents the probabilistic contribution. Moreover, the notion of line of protection, which allows merging the contribution of several provisions to achieve a common mission, asks for specific probabilistic support to ensure that the reliability targets are effectively met, for a given level of the DiD, by the line of protection as a whole (i.e. jointly by all the provisions of the line of protection). Although reliability data and initiating event frequencies on SFRs are not sufficient, probabilistic safety assessment should be extremely beneficial for systematically comprehending the risk characteristics of a plant with respect to a risk potential. Design improvement can be effectively made in such a way as to control and minimize the risk. With regard to safety assessment, PIEs are selected and quantified with respect to their occurrence frequencies. On the basis of the frequencies, PIEs are classified into the appropriate event categories and assessed in order to determine the safety criteria with respect to the event category.

5.4. International cooperation and harmonization of safety requirements

Although considerable licensing experience has already been gained in licensing SFRs, it is less than that for LWRs and the safety objectives and safety approaches evolve with time. The new generation SFRs will have different design features, introducing innovative technology to improve plant performance. These efforts will be expected to facilitate increased cooperation and to establish common safety requirements to enhance the safety of advanced SFRs.

International forums such as the Generation IV International Forum and INPRO provide a framework for international cooperation organized to carry out the R&D needed to establish advanced and/or innovative technology meeting the next generation reactor requirements. In the safety area of R&D, experiments and analytical model development cooperation are being carried out on passive safety and severe accident issues.

It is expected that the licensing of advanced SFR designs in different countries will be facilitated through the sharing of resources and knowledge of the national regulatory process. Moreover, it is desirable to converge safety requiremens among countries. In order to achieve the long term objective of establishing an international standard of safety requirements, interaction with the safety authority is useful for the development of the advanced SFR, where experience and practice are limited in licence procedures. Pre-application to perform a safety review would be an efficient approach to use in advance of actual licensing.

6. CONCLUSIONS

This paper summarizes the safety characteristics of SFRs and the safety approach taken in those SFRs which were either planned or actually designed, constructed and operated. These experiences show that SFR technology has matured well and to a level that such a reactor concept is licensable. However, further development for the commercialization of SFRs is still necessary to meet the requirements of the next generation of reactors.

The concept of DiD will be applied to the safety design of advanced SFRs. The safety level can be further improved, especially enhancing prevention and mitigation features, with more emphasis on passive safety features. Through prevention, detection and control of accidents, CDAs will be excluded from DBEs. The most safety significant issue for SFRs has been, and will continue to be, CDAs which might lead to severe recriticality owing to SFR characteristics. As regards commercialization of SFRs, not only prevention but also mitigation of typical severe core damage needs to be enhanced, taking into account the increase in the number of plants and their scale. In particular, the safety approach with elimination of severe recriticality is highly desirable and will contribute to establishing public acceptance of SFRs.

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STRUCTURAL MATERIALS: NEW CHALLENGES, MANUFACTURING AND PERFORMANCE

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Abstract

Important criteria of innovative fast reactors and advanced fuel cycle initiatives are improved efficiency, economic competitiveness and reduction of waste. To reach these goals and keep high safety standards, at least at the level of currently operating nuclear reactors, key issues are the availability of suitable structural materials and their performance assessment. The authors, on the basis of the wealth of experience gained in the European Union, India and Japan, aim to define the challenges and current status of material development and set the agenda for R&D in the coming years and decades. It is hoped that the joint perspective would enable realizing the expected criteria of sustainability envisaged through sodium cooled fast reactors and closed fuel cycles.

1. INTRODUCTION

The authors, employing the wealth of experience gained by the European Union (particularly within the Euratom framework programmes), India and Japan aim to define the challenges and the current state of the art and set the agenda for R&D in the coming years and decades. This paper reviews international perspectives on materials, manufacturing and performance on austenitic stainless steels, 9–12Cr ferritic–martensitic (F/M) steels and oxide dispersion strengthened (ODS) steels which are the current and prospective structural materials for fast spectrum reactors. The paper gives a joint perspective that would enable realizing the expected criteria of sustainability envisaged through sodium cooled fast reactors (SFRs) and closed fuel cycles.

It is recognized that fast reactors employing the closed fuel cycle will play an eminent and major role in realizing energy sustainability. Large scale exploitation of fast reactors will require meeting sustainability requirements such as economic competitiveness, safe and optimized waste management, increased proliferation resistance, improved use of uranium and thorium and enhanced efficiency. The designers of fast reactors, along with material specialists, have a key role in meeting the above mentioned criteria of sustainability.

Fast neutrons which have lower cross-sections for fission demand an increase in neutron flux ($\sim10^{15}$ $n\cdot cm^{-2}\cdot s^{-1}$) by an order of magnitude over the thermal reactors to achieve the desired linear heat rates. The core materials are, therefore, subjected to a demanding environment of high fast neutron flux coupled with high temperatures and high thermal gradients due to the high heat transfer property of sodium which allows it to extract heat efficiently (sodium is the preferred choice from the efficient heat removal point of view, under both steady and transient conditions). From the design point of view, the core components (e.g. fuel subassemblies, cladding and wrapper tubes) of most innovative systems must, during operation, withstand high levels of radiation damage, high temperatures, complex thermomechanical loading and corrosive/erosive effects due to the flowing coolants. Consequently, new issues and challenges related to the development and qualification of structural materials for core components and primary system circuits must be evaluated to ensure safe and reliable operation of these systems.

In Europe, the sustainability approach, as indicated by the Sustainable Nuclear Energy Technology Platform, indicates that the SFR is the primary technological choice for development. However, a second technological route (gas cooled fast reactor or lead cooled fast reactor) should be evaluated for selection by 2012. Moreover, Europe will assess as well the option to transmute nuclear waste with external neutron driven subcritical systems. For the range of service conditions expected for these Generation IV and advanced transmutation systems, including possible accident scenarios, sufficient data must be available to demonstrate that the candidate materials for reactor core and primary coolant components meet the design objectives.

A high flux of fast neutrons induces atomic displacements in the core structural materials leading to phase instabilities, void swelling, irradiation creep and changes in mechanical properties. These phenomena are interlinked and it has been shown that void swelling is sensitive to the evolution of phases in austenitic stainless steels and has the dominant influence on irradiation creep behaviour, mechanical strength and ductility. Variations in chemical composition and microstructure influence void swelling and irradiation creep. Thus, the solution to this major challenge is judicious choice of chemical composition and tailoring of microstructures.

Void swelling, irradiation creep and irradiation embrittlement arising out of fast neutron exposure of core structural materials are important phenomena that determine the residence time of fuel elements in the core of fast spectrum reactors. For economic viability, the target burnup required for fast spectrum reactors is more than 20 at.% of heavy metal (200 000 MW·d/t), and this can be achieved only by the use of materials resistant to void swelling, irradiation creep and irradiation embrittlement, as well as satisfying the high temperature mechanical properties. Since fuel cycle cost is strongly linked with burnup, selection of materials resistant to void swelling and irradiation creep is an important research endeavour. Design studies for large and medium scale SFRs set target discharge average burnup to be 150 GW·d/t. In order to improve the thermal efficiency of the plants, the maximum outlet coolant temperature at the reactor vessel is determined to be 823 K, and corresponding maximum ('hot spot') temperatures of the cladding tube and duct tube are 973 K and 853 K, respectively. Improved varieties of 316 austenitic stainless steels and oxide dispersion (F/M) steels emerge as front runners in meeting these requirements.

The developments in Ti modified 316 austenitic stainless steels give enough confidence to take oxide fuels to burnup of 120 000 MW·d/t. For doses above 120 dpa, austenitic stainless steels are not employed, as void swelling is found to be substantial. Though F/M steels such as modified 9Cr–1Mo and HT9 exhibit higher void swelling resistance than conventionally used austenitic stainless steels, these alloys display poor thermal creep strengths at temperatures above 923 K. This has led to restrictions on achieving high burnup of fuel with clad operating at temperatures in the range 870–970 K. However, ODS is a promising means of extending the creep resistance of F/M steels beyond 973 K without sacrificing their inherent advantages of high thermal conductivity and low swelling. It is inferred that the target burnup of 250 000 MW·d/t would only be achieved with ODS iron–chromium base steels. In the following paragraphs, details of selected research that has enabled development of the above high performance materials are highlighted. The R&D of modified austenitic stainless steels and ODS alloys in Europe, India and Japan are described.

In Europe, the European Commission is supporting the Euroatom FP7 Generation IV and Transmutation Materials (GETMAT) project [1]. The GETMAT project, being of cross-cutting nature, addresses structural materials for core components and the primary systems of fast neutron nuclear devices cooled with Na, Pb/Pb–Bi eutectic and He. Thus, the objectives of GETMAT are to contribute to the development qualification and ranking of two classes of alloys, i.e. ODS steels and F/M steels. The experimental activities are complemented with the development of physical models with the aim of understanding and improving the predictability of the materials' performance. Moreover, the GETMAT project aims as well to streamline and integrate, in a comprehensive way, the R&D effort of the European materials laboratories for a wide ranging study of the performance of the two classes of alloys. The main objectives of the GETMAT project are: (i) improvement and extension of knowledge on 9–12Cr F/M steels, (ii) ODS alloy development and characterization, (iii) joining and welding procedures qualification, (iv) development and definition of corrosion protection barriers, and (v) improved modelling and experimental validation.

In Japan, the R&D programme on materials for fast breeder reactors (FBRs) is being conducted as a part of the Fast Reactor Cycle Technology Development project (FaCT project) led by the Japan Atomic Energy Agency (JAEA). The FaCT project is for the commercialization of the Japanese sodium cooled fast breeder reactor (JSFR) by around 2050 and for operation of a JSFR demonstration reactor by around 2025. The JAEA has selected ODS and precipitation hardened ferritic steels as the most prospective candidate materials for cladding and duct tubes, respectively. For the structural materials of the JSFR, 316FR and modified 9Cr–1Mo steels are to be applied [2]. The 316FR is a low carbon and nitrogen added stainless steel that has been developed in Japan to improve high temperature creep properties [3]. The chemical compositions of the steel are optimized within the specifications of SUS316 in the Japanese Industrial Standard. A unique feature with regard to the chemical composition is that phosphorus is also added. This material has been adapted to the intermediate heat exchanger of the JOYO experimental fast reactor located in Oarai, Japan, and will be used for the reactor vessels and internal structures of the JSFR. Modified 9Cr–1Mo steel is to be used for primary and secondary coolant systems, intermediate heat exchangers and steam generators to take advantage of the low thermal expansion and good elevated temperature properties of this material. Alloy development of these steels has almost been completed, but there are some important issues to be addressed. The first one is acquisition of long term data that form the basis for a 60 year design and the development of evaluation methods of material performance such as very long term creep fatigue. The second point is the development of manufacturing technology to produce components required for the JSFR, such as the large diameter forged ring for the reactor vessel (316FR), the very thick forged plate and the very long and thin double walled heat exchanger pipes (modified 9Cr–1Mo steel). These are necessary to pursue the economic advantages of the plant.

The final point to be noted is that these steels have not been registered in current Japanese nuclear codes and standards published by the Japan Society of Mechanical Engineers (JSME) which will be used for the design and construction of the JSFR. Therefore, it is necessary to register the steels in the JSME code [4] and to standardize the allowable stresses permitted for the JSFR structural design at elevated temperature. The JSME code for the 2016 edition is scheduled to be used for the licensing process of the JSFR demonstration plant, operation of which is envisioned to start around 2025.

The development of advanced materials for sodium cooled fast spectrum reactors in India aimed at extending the life of reactors beyond 60 years and the burnup of fuel to 200 000 MW·d/t has recently been reviewed [5, 6]. The roadmap of materials development is linked to the choice of fuel, i.e. oxide, carbide, metallic with or without minor actinides. For oxide fuelled FBRs, 15Cr–15Ni–2.3Mo and Ti additions (alloy D9 and its variants with P additions) is currently the material for both clad and wrapper. For burnup exceeding 120 dpa, 9Cr–1Mo as wrapper and ODS steels as cladding material are being developed. In order to reduce susceptibility of welds to corrosion and stress corrosion cracking, low carbon stainless steel grades 304 and 316 strengthened by nitrogen alloying and termed 304LN and 316LN have been selected for the prototype FBR (PFBR) out-of-core structural components. Periodic, non-destructive inspection of reactor core and structural components is essential for early detection of material degradation processes such as intergranular corrosion, stress corrosion cracking, pitting corrosion, creep and fatigue damage either on-line or during shutdown. Innovative ultrasonic and eddy current non-destructive evaluation techniques have been developed for inspection of structural and core components of the reactor and steam generator components [5].

2. AUSTENITIC STAINLESS STEELS

The performance of austenitic stainless steels with the exception of 321 has been satisfactory in fast reactors. Grades with which good performance has been achieved include 304, 304LN, 316, 316L, 316LN and 316FR. There have been a number of cracks and sodium leaks associated with stainless steel 321 welds in Phenix secondary sodium piping and steam generators, and superheater and reheater vessel shells of the PFBR. The cracks are attributed to delayed reheat cracking. As a result, 321 has been gradually replaced by 316LN in Phenix. In view of this experience, stabilized grades 321 and 347 will not be considered for future fast reactors. Austenitic stainless steels are widely used as structural materials for in-core and out-of-core components. For oxide fuelled FBRs in India, 15Cr–15Ni–2.3Mo with Ti additions (alloy D9 and its variants with P additions) is currently the material for both clad and wrapper. Type 316FR has been developed in Japan [3]. To improve the high temperature strength, the chemical compositions of the steel are optimized within the specifications of SUS316 in the Japanese Industrial Standard.

2.1. Core materials

Structural materials for fast reactor core components have evolved continuously over the years, resulting in substantial improvement in fuel performance. The first generation materials belonged to austenitic stainless steel types 304 and 316. A 20% cold welded 316 stainless steel has been used in the clad and wrapper of the fast breeder test reactor and TEM analysis of its irradiated wrapper shows the presence of voids beyond 40 dpa [7]. These steels quickly reach their limits because of unacceptable swelling at doses higher than about 50 dpa. For the development of swell resistant alloys, it has been found that it is necessary to optimize the composition of the minor alloying elements such as titanium, silicon and phosphorus, which have a major influence on swelling. This optimization is brought about with a view to introducing microstructures which are designed to minimize degradation of certain properties during the irradiation. Many improvements were made by changing the concentration of the major and minor elements, as well as by modifying the microstructures by introducing cold work. This has led to the development of advanced core structural materials such as alloy D9 (15Cr15Ni–Ti modified steel) for which the incubation dose for swelling is improved compared with 316. Alloy D9 has been designated as a core structural material in the 500 MW(e) PFBR under construction in India. Further, high power fast reactors will require materials better than alloy D9 for higher burnup. Efforts are under way to develop improved versions of D9, notably by modifying the composition of minor elements, namely, silicon and phosphorus. For eventual fulfilment of all the requirements of the design engineer, issues concerned with metal joining and component qualification are being addressed.

2.1.1. Creep properties of austenitic stainless steels

It has been noted that the creep rupture strength of alloy D9 is better than that of 316 by a factor of four at 923 K, by a factor of about six at 973 K and by about a factor of ten at 1023 K. The improvement in strength is found to be the consequence of prolonged secondary creep exhibited by alloy D9. Austenitic stainless steels derive their strength from solid solution strengthening and from carbide precipitation in the matrix. In the case of 316, fine $M_{23}C_6$ types of carbide are known to form at 873 K. At 973 K and above, coarsening of carbides takes place, enabling recovery and thus decreasing the efficiency of precipitation strengthening. In alloy D9, carbon is partitioned between matrix titanium and other alloying elements such chromium. The fine (a few nanometres) secondary titanium carbides form predominantly in the matrix, imparting higher creep rupture strength.

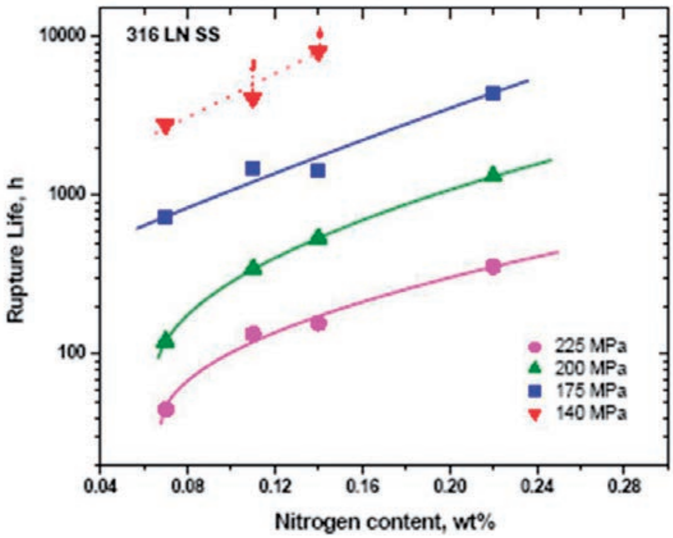

FIG. 1. Effect of nitrogen on creep rupture strength of 316LN.

2.2. Out-of-core components

2.2.1. Development of nitrogen added type 316L

Increasing the lifetime of the reactor to 60–100 years is under consideration in order to reduce the cost of nuclear energy. This necessitates reliable creep data generation at various heats for structural materials and establishing appropriate life prediction techniques based on knowledge of prevailing deformation, damage and fracture mechanisms. Significant heat to heat variation in the creep rupture properties has been observed in spite of strict control of chemistry, grain size and other processing parameters during manufacture of 316 stainless steel. Heat to heat variations have been attributed to differences in grain sizes and amounts of minor elements such as carbon, boron and nitrogen in the different heats of the material.

An understanding of the microstructural changes, dislocation evolution and damage mechanisms during long term deformation in various heats has enabled development of robust creep life prediction models and prediction of lives under service conditions that are not covered by laboratory testing. These studies also became indispensable in the development of nitrogen alloyed 316 that possesses higher creep resistance compared with 316 (Fig. 1). A basic mechanistic understanding of the evolution of creep induced microstructural changes in 316LN welds and weldments has revealed that the creep design of welded components has to be carried out on the basis of the properties of the weld joint rather than that of the weld metal. Weld strength reduction factors have been developed for 316LN weld joints for various temperature and stress combinations in the design of FBR components.

2.2.2. Material development of 316FR

Type 316FR is a low carbon and nitrogen added stainless steel that has been developed in Japan to improve high temperature creep properties [3]. The chemical compositions of the steel are optimized within the specifications of SUS316 in the Japanese Industrial Standard. A unique feature regarding the chemical composition is that phosphorus is also added. The chemical composition of 316FR is given in Table 1. This material has been adapted to the intermediate heat exchanger of the JOYO experimental fast reactor located in Oarai, Japan, and will be used for the reactor vessels and internal structures of the JSFR.

2.2.2.1. Acquisition of long term data and development of evaluation methods

Basic material data such as tensile, creep, fatigue and creep fatigue data have been obtained for 316FR and a draft material strength standard for this material has been summarized for a 40 year design as the demonstration reactor design standard (DDS). However, the design life of the JSFR is 60 years and data gathered over a longer period are required and existing procedures for material properties such as creep fatigue interaction should be re-evaluated from the viewpoint of their applicability to a 60 year design. Therefore, in the case of creep tests for example, tests of up to 200 000 h and more are being planned. Such long term tests will be continued after the start of the operation of the JSFR's demonstration plant and the data will be used for the validation and improvement (margin optimization) of the material strength standard. As for the evaluation technologies of material properties, evaluation of long term creep fatigue life, including that of welded joints, will be of prime importance. Those procedures are also to be included in the JSME code, which will be published in 2016 and used for the licence process.

TABLE 1. CHEMICAL COMPOSITION OF 316FR IN THE DDS

C	Si	Mn	P	S	Ni	Cr	Mo	Al	N
≤0.020	≤1.00	≤2.00	0.020–0.045	≤0.030	10.00–14.00	16.00–18.00	2.00–3.00	≤0.05	0.06–0.12

2.2.2.2. Fabrication technologies

The diameter of the reactor vessel of the commercialized plant of the JSFR would be around 10 m. The outlet temperature of sodium is 550° C. From the viewpoint of structural simplicity that will lead to enhanced economic benefits, the hot vessel concept is being explored in the FaCT project. In this case, to ensure the structural integrity of the reactor vessel, a forged ring is desirable in the vicinity of the sodium surface where the largest stress intensity induced by the movement of the sodium surface is expected. Therefore, technologies for fabrication of such a forged ring with a large diameter are to be investigated. The main issues would be achieving chemical compositions and maintaining material strength (particularly short term properties) for such a large structure. The latter could be an issue because the grain size might be larger than usual owing to restrictions on the force used for forging.

2.2.2.3. Codification in the JSME code

The scope of the DDS document included a 40 year design and involved various allowable stresses for elevated temperature design. The effects of neutron irradiation and the liquid sodium environment on these allowable limits were also given. The allowable stresses were applicable to all product forms, except extremely thick forgings, i.e. greater than 220 mm. They were determined from material test data that were generated in Japan. For example, S_u and S_y were determined on the basis of the lower bound of 99% reliability level against failure and yielding, respectively, using a number of tensile test data. In addition, the DDS document also represented material characteristic equations such as the creep rupture equation, the creep strain equation, the equation for 'best fit' curve for low cycle fatigue life and so on. These equations were required for the structural design assessment at elevated temperatures, as well as for the determination of allowable stresses. Since 316FR is to be used as a structural material for the JSFR, the material strength standard for the steel must be established prior to the beginning of the licensing process. The material strength standard for elevated temperature design as well as elevated temperature methodologies for the 60 year design of the JSFR will be published in 2016 by the JSME.

3. F/M STEELS

In the long term, F/M steels (9–12%Cr) have been identified as the core component structural materials, owing to the inherent low swelling behaviour of bcc ferrite. However, the increase in the ductile to brittle transition temperature

(DBTT) due to irradiation is a cause for serious concern in the use of ferritic steels. Several steps have been taken to improve the performance of the steel with respect to high temperature creep strength and embrittlement problems. This includes modification of the steel through alloying additions, control of tramp elements and tailoring of microstructure through metallurgical treatments. The upper shelf energy and shift in DBTT saturate at irradiation doses of about 20 dpa. Increase in toughness has been obtained by ensuring a fully martensitic structure, avoiding formation of delta ferrite in 12Cr steels by suitable chemistry. Refining the prior austenite grain size by optimizing the austenitizing temperature and tempering treatments are methods used to reduce the strength and improve the toughness of the 9–12%Cr steels. The 9Cr–1Mo grades of ferritic steels are reported to show the lowest increase in DBTT among the ferritic grades [6]. Hence, the 9Cr–1Mo class of steel is being considered for the wrapper in future FBRs. At present, the indicated reference structural materials for applications to core and primary components of the different fast reactor and transmutation systems (GETMAT project), belong to the same classes, i.e. the 9–12Cr F/M steels for use up to 600°C and the ODS high Cr steels for temperatures beyond 600°C.

Modified 9Cr–1Mo steel was originally developed to replace conventional austenitic stainless steels in the major components of SFRs. The steel has been standardized as Grade 91. However, the specifications such as chemical compositions and heat treatment conditions should be revisited from the viewpoint of applicability to the 60 year design for SFRs.

3.1. Improvement and extension of 9–12Cr F/M steels qualification

The European approach has considered conventional grade 9–12Cr F/M steels that are produced on an industrial scale. Although industrial experience exists, more comprehensive validated data on the 9–12Cr F/M steels are still needed to qualify their use in the in-service conditions planned for each specific fast neutron nuclear system. In particular, the database of 9Cr1MoVNb (T91) steel, the reference for different nuclear applications, needs to be completed. Within the GETMAT project, the objective to improve and extend the 9–12Cr F/M steel qualification will be reached through the generation of mechanical, microstructural and metallurgical data from post-irradiation experiments (Table 2). The combination of results thus generated will facilitate the performance assessment (300–570°C and up to ~70 dpa) and the study of the combined effects of the steels in an irradiation field and in contact with different coolants and even under flowing conditions. An understanding of the 9–12Cr steels in the above mentioned conditions is of paramount importance to support,

TABLE 2. IRRADIATION CONDITIONS AND THE MECHANICAL TESTS/EXAMINATIONS FORSEEN IN THE IRRADIATION STUDIES OF THE GETMAT PROJECT

	Reactor					
Experiment	MATRIX Phenix	STIP 4–5 SINQ	MEGAPIE	ASTIR BR2	LEXUR II BOR60	IBIS and SUMO HFR
Spectra	Fast neutrons	High energy protons and neutrons	High energy protons and neutrons	Thermal and fast neutrons	Fast neutrons	Thermal and fast neutrons
Materials	T91, T92, EUROFER T91,T92 coated ODS (9–20Cr)	9–12 Cr welds ODS (9–20Cr)	T91 316L	T91, T91 coated, welds, ODS	T91, T91 coated, welds, 316L	T91, Eur–ODS T91 coated, 316L, welds
Tests, examinations	Tensile, impact, CT, fractog., SEM, TEM	Tensile, bending, Charpy, SPT, TEM	Tensile, bending, SPT, SIMS, XPS XRD, SEM, TEM	Pressurized tubes CT, tensile, Charpy	Tensile, corrosion	Tensile, KLST, SEM, TEM
Irradiation temperature	390–530°C	300–700°C	250–375°C (T91 beam window)	350°C, 450°C	400°C: 316L 480°C and 550°C: T91	300°C, 500°C
Dose range	30–65 dpa	10–20 dpa	~7 dpa (T91 beam window)	~5 dpa	Up to 20 dpa	2 dpa
Environment	Na	Inert gas	Pb–Bi	Pb–Bi, inert gas	Pb, inert gas	Pb–Bi Na (SUMO)

wherever possible, technical solutions for specific design items (e.g. clad, wrapper)

In the FaCT project, R&D on modified 9Cr–1Mo steel is also being actively pursued in order to evaluate the material for use in the primary and secondary coolant loops, intermediate heat exchangers and steam generators. The material is basically equivalent to ASME Grade 91, but optimization of specifications for its application to the JSFR is envisaged as enhancing resistance in the temperature ranges and environment encountered in liquid metal FBRs.

3.2. Acquisition of long term data and development of evaluation methods

As is the case for 316FR, basic material data on modified 9Cr–1Mo steel have been obtained and a draft material strength standard of this material has been summarized for the 40 year design as DDS. However, the design life adopted in the FaCT project is 60 years and data gathered over a longer period are required and existing procedures for material properties such as creep fatigue interaction should be re-evaluated from the viewpoint of the 60 year design. Therefore, in the case of creep tests for example, those up to 200 000 h and more are being planned. Long term tests will be continued after the start of the operation of the JSFR demonstration plant and the data will be used for the validation and improvement (margin optimization) of the plant. As for the evaluation technologies of material properties, evaluation of long term creep fatigue life, including that of welded joints, will be of primary importance. In the case of modified 9Cr–1Mo steel, focus should be on the evaluation procedure of creep fatigue evaluation, taking type IV damage into account.

3.3. Fabrication and codification

In the FaCT project, for the construction of the JSFR, manufacturability of some special form components, such as thick forgings for steam generator tube sheets, thin walled seamless pipes of large diameter and thin walled steam generator tubes of small diameter, should be investigated. Allowable stresses for these materials specific to the JSFR will also be codified in the JSME code which will be published in 2016. Figure 2 shows a thick ingot for forging being prepared for evaluation of material properties.

Figure 3 indicates the tensile properties of this material obtained by material experiments and used for the determination of allowable stresses. Figure 4 shows creep data for this material. A number of data whose rupture time is over 100 000 h have been already obtained and long term tests are also going on. The JSME code, which will be published in 2016, will also represent material

FIG. 2. An ingot for thick forging.

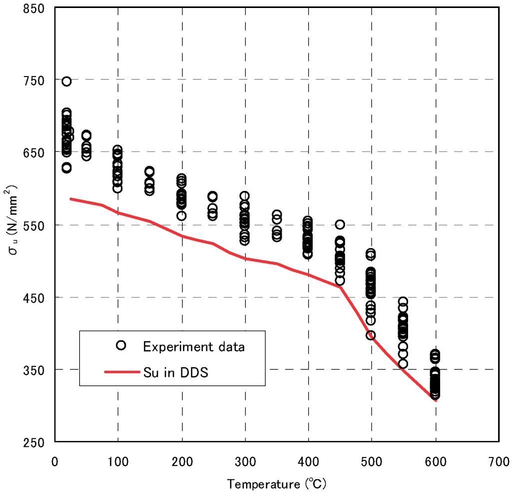

FIG. 3. Tensile strength of modified 9Cr–1Mo steel.

characteristic equations such as creep rupture equation, creep strain equation, equation of best fit curve for low cycle fatigue life and so on. These equations were required for the structural design assessment at elevated temperatures, as well as for the determination of allowable limits.

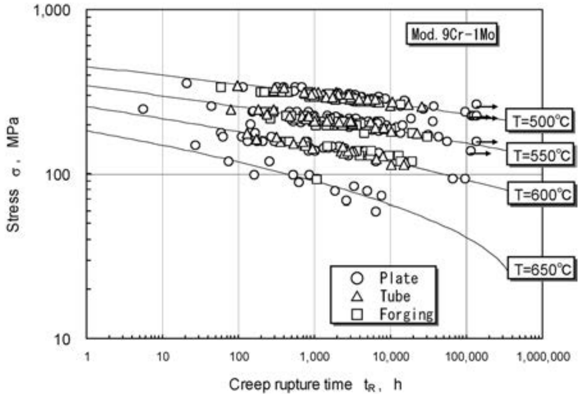

FIG. 4. Creep rupture curves of modified 9Cr–1Mo steel.

3.4. Type IV cracking ferritic steel weld joints

The creep strength of the ferritic steel weld joint is lower than that of the weld metal because of type IV cracking, which is characterized by creep cavitation and fracture in the fine grained/intercritical heat affected zone of the weld joints. Creep studies on simulated heat affected zone specimens have shown that the creep strength of these zones is lower than that of the base metal, weld metal and coarse grained heat affected zone. In the case of weld joint, deformation of the weak fine grained/intercritical heat affected zone is constricted during creep by relatively strong base metal on one side and the coarse grained heat affected zone on the other side, resulting in a triaxial stress state which results in cavitation and final fracture in this zone with very little ductility. Although it is difficult to avoid type IV cracking, several methods are being adopted to improve type IV cracking resistance. Strength homogeneity across the weld joint can be improved by normalizing the component after welding [7].

3.4.1. Dissimilar weld joints

Dissimilar metal welds between austenitic stainless steel and Cr–Mo ferritic steels are prone to extensive service failures due to large thermal stresses induced at the weld fusion line because of the different coefficients of thermal expansion of the ferritic steel base metal and the weld metal. A unique trimetallic transition joint between austenitic stainless steel and Cr–Mo steel, with an intermediate alloy 800 piece, has been designed, developed and characterized [7]. The trimetallic transition joint, 316LN/alloy 800/modified 9Cr–1Mo steel adopted for the steam generator circuit of the 500 MW(e) PFBR has shown a factor of four improvement in life under thermal cycling conditions. The long range diffusion of carbon during high temperature exposure of dissimilar ferritic joints results in the formation of hard and soft zones in the weldments. The driving force for diffusion of carbon is shown to be the chemical potential gradient across the joints. The formation of these zones leads to degradation of mechanical properties and shorter endurance of the welded component. Use of diffusion interlayers as barriers between the joints could prevent the formation of a deleterious zone at the weld interface. The optimum thickness of the Inconel interlayer between 9Cr–1Mo and 2¼Cr–1Mo joints was found to be about 80 μm. Further, theoretical calculations showed that copper and cobalt could also act as effective diffusion barriers similar to the nickel based Inconel 182 interlayer [7].

4. PRECIPITATION HARDENED FERRITIC STEEL

4.1. Material development

Since the maximum temperature of the duct tubes is expected to be about 873 K in the FaCT project, precipitation hardened ferritic steels can be applicable to the JSFR. For the duct tubes, PNC-FMS was selected as the primary candidate material at the beginning of a project performed prior to the FaCT project. Since 1983, Japan has been developing PNC-FMS, whose chemical composition is defined as Fe–0.12C–11Cr–0.5Mo–2W–0.4Ni–0.2V–0.05Nb–0.05N in mass per cent. The tentative Material Strength Standards (MSS) for the PNC-FMS were established by 1993 and irradiation tests in the JOYO facility and the Fast Flux Test Facility were conducted to prove its validity. Moreover, PNC-FMS was applied to the duct tube of reflectors in JOYO MK-III cores. The core support structure of the JSFR will be made from austenitic stainless steels (SUS316), and therefore the entrance nozzle of every fuel subassembly should be SUS316. Design studies on the JSFR cores in the FaCT project consider that a PNC-FMS duct tube will be joined with the SUS316 duct tubes at their lower and upper ends.

4.2. Fabrication and verification

There are two methods of joining PNC-FMS to SUS316 to manufacture duct tubes. One is a mechanical joint with a screw thread and the other is dissimilar welding. Mechanical joints can endure thermal stress due to differences in the thermal expansion coefficient, but high pressure flowing coolant will leak through their gaps. Therefore, dissimilar welding is a favoured way to avoid such coolant leakage. In the FaCT project, three dissimilar welded duct tubes were initially manufactured. Two duct tubes were manufactured by welding hexagonal PNC-FMS tube to hexagonal SUS316 tubes by either electron beam or tungsten inert gas arc (TIG) welding. The third one was manufactured by cold drawing a TIG welded circular tube into a hexagonal tube. The dimensional inspections of the three tubes showed little difference. The hexagonal tube to tube method with electron beam or TIG welding appears to be more promising, because plastic deformation will induce detrimental effects to the welded part during the cold rolling process.

Trial manufacturing of dissimilar welded duct tubes began in 2007 in order to establish the manufacturing technology for mass production by 2010 and to provide for fuel pin bundle irradiation testing in the JOYO facility.

To evaluate the integrity of dissimilar welded parts, tensile specimens were machined and then tested. Fractures occurred in SUS316 base metal from room temperature to 873 K, and in PNC-FMS base metal up to 973 K.

The material irradiation test of dissimilar welded specimens started using the Core Material Irradiation Rig in the JOYO facility from 2006, in order to evaluate the irradiation effects on mechanical properties. The tentative MSS for the PNC-FMS were established by 1993, and irradiation tests in the JOYO facility and the Fast Flux Test Facility have been conducted to prove its validity.

5. ODS ALLOYS

The F/M alloys (9–12%Cr) exhibit higher void swelling resistance than conventionally used austenitic stainless steels under irradiation. These steels are suitable for a variety of coolant and tritium breeding options and have good mechanical strength only up to 773–823 K. This disadvantage of low mechanical strength at high temperatures can be overcome by introducing thermally stable oxide dispersions in the ferrite or F/M matrix. Yttria (Y_2O_3) is the main oxide additive used. Figure 5 shows the TEM micrograph of Fe–9Cr–0.11C–2W–0.2Ti–0.35Y_2O_3 F/M ODS alloy showing yttria particles. The dispersed fine Y_2O_3 particles improve high temperature strength by blocking mobile dislocations and retard irradiation swelling by acting as trapping sites for point defects induced by irradiation. These ODS steels may enable even higher service temperatures up to 923–1073 K in SFR applications. However, it is evident that the fabrication of ODS alloys at an industrial level is one of the most important challenges to be faced for a sustainable development of innovative fast reactors.

Within the GETMAT work programme, a preliminary assessment of 9Cr and 14Cr ODS alloys is required to qualify the two selected fabrication routes,

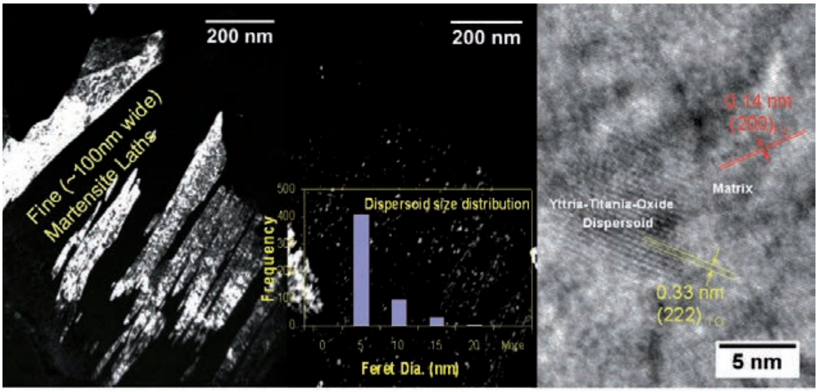

FIG. 5. TEM micrograph of Fe–9Cr–0.11C–2W–0.2Ti–0.35Y_2O_3 F/M ODS alloy developed in India.

i.e. powder metallurgy and an innovative casting method. The test matrix defined for the characterization of the three ODS alloys includes high temperature mechanical tests (creep, fatigue, etc.) in an inert environment and with different coolants, as well as compatibility tests. Moreover, post-irradiation experiments of ODS alloys irradiated in the Phenix, HFR and SINQ will complement the set of data which will be available to aid the understanding of the ODS alloys under specific irradiation conditions. The final objective of this part of the GETMAT project is to perform a preliminary assessment of these alloys for the applications currently envisaged in the different reactor systems, but more generally to pave the way for the future of ODS alloy development in Europe for nuclear applications.

Low fracture toughness, high DBTT and the finding of suitable welding processes (fusion welding is undesirable) for ODS alloys continue to be causes of concern. A low defect tolerance exists in ODS alloy components due to high DBTT and low fracture toughness. In the case of mass production of ODS alloy clad tubes and plates of blanket systems, the initial billets are generally produced by hot isostatic pressing of mechanically alloyed powders. In hot isostatic pressing, the impurity particles present on powder particles become embedded on grain boundaries or form agglomerated particles, leading to accelerated creep cavity formation and associated reduction in life. Though the creep resistance of ODS alloys could be improved marginally by choosing coarse particles initially, there could be other modifications needed in powder production and processing conditions to eliminate prior particle boundaries. Much more work is still needed on ODS steels before they can be used in the critical structural components of fission and fusion systems.

In the JAEA, two types of ODS steel have been developed for the cladding tubes which have been durable at high temperature and which have endured in a high neutron dose environment since 1987. One is 9Cr–ODS steel aiming at higher radiation resistance with a basic chemical composition of Fe–0.13C–9Cr–2W–0.2Ti–0.35Y_2O_3 mass per cent, and the other is 12Cr–ODS steel aiming at higher corrosion resistance with a basic chemical composition of Fe–0.03C–12Cr–2W–0.3Ti–0.23Y_2O_3 mass per cent. On the basis of the results, it was decided from the viewpoints of formability and irradiation performance and the like that 9Cr–ODS steel would be the primary candidate and 12Cr–ODS steel the secondary [8], and that manufacturing technology development for mass production will be performed only on 9Cr–ODS steel. R&D on 12Cr–ODS steel is to be performed as this is the backup material.

5.1. Fabrication and characterization

The manufacturing process of the ODS steel cladding tubes in the FaCT project is represented in Fig. 6. Argon gas atomized pre-alloy powders with Y_2O_3 particulates were mechanically alloyed in an argon gas atmosphere by using a 10 kg attriter. The mechanically alloyed powders were then loaded into mild steel cans, degassed and hot extruded at 1423 K. The extruded bars were machined and drilled to mother tubes. The ODS steel cladding tubes of 8.5 mm diameter, 7.5 mm inner diameter and 2 m in length were manufactured through repeating four times a process of cold rolling with a reduction rate of near 50%. Intermediate heat treatments during the cold rolling process and final heat treatment introduce phase transformations from α to γ in 9Cr–ODS steel and recrystallization in 12Cr–ODS steel to reduce strength anisotropy by grain morphology control [9, 10]. The ultrasonic C-scan imaging technique has been established for evolving stringent quality assurance procedures that ensure manufacturing quality. Mechanical alloying and hot consolidation processes dominate the dispersoid morphology and resultant mechanical properties.

In order to evaluate the mechanical properties of manufactured ODS steel cladding tubes, ring tensile tests and creep rupture tests with internal pressure

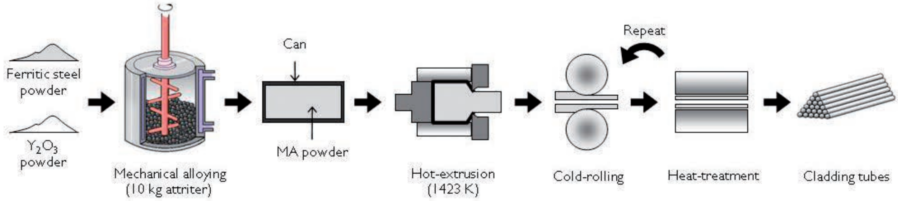

FIG. 6. Manufacturing process of ODS steel cladding tubes.

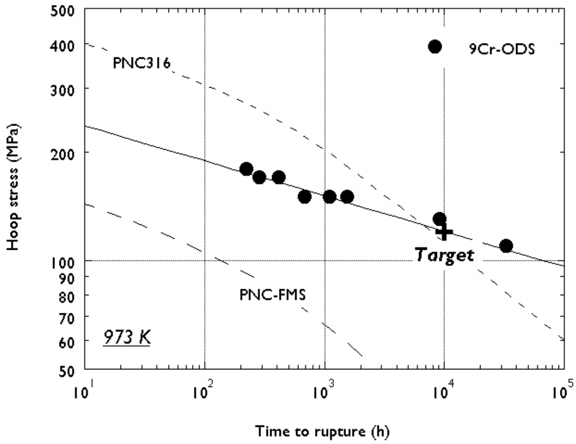

FIG. 7. Creep rupture strength of manufactured 9Cr–ODS steel cladding tubes.

were carried out. Manufactured ODS steel claddings showed improved tensile strength over the entire temperature region and the uniform elongation was adequately maintained. Figure 7 shows the internal creep rupture strength in comparison to PNC-FMS and conventional austenitic steel (PNC316). The internal creep rupture strength level attained 120 MPa for 10 000 h at 973 K, which is a target required by the JSFR fuel design. This strength level is much better than that of PNC-FMS and superior even to PNC316.

Sodium environmental effects on ODS steel claddings were investigated by corrosion and mechanical strength tests at elevated temperature [11]. For PNC-FMS, a clear strength reduction occurred above 873 K due to decarburization into sodium. While decarburization was observed, the ODS steel did not show such a clear strength degradation, in contrast to conventional steel, even at high temperatures and after long term exposures. This suggested that the fine Y_2O_3 particles remained stable in ODS steels and that the strengthening mechanism of the steel was maintained. Figure 8 shows the results of creep rupture tests with internally pressurized tube specimens under stagnant sodium immersion. Creep rupture strength in sodium is equal to the strength in air. This result shows excellent stability up to 973 K under stagnant sodium conditions, and the effect of sodium on creep rupture properties is negligible in practical application.

On the basis of these results, the MSS for the fuel pin irradiation test in the JOYO facility was tentatively compiled in 2005. Moreover, the long term stagnant sodium immersion and creep rupture tests with pressurized tube

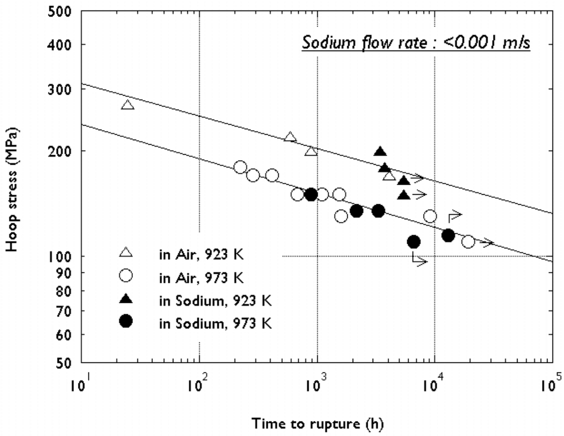

FIG. 8. Creep rupture strength of manufactured 9Cr–ODS steel cladding tubes in sodium.

specimens in air and stagnant sodium are being continued and the results will be reflected in the upgrade of the MSS.

To establish the MSS for fuel pin mechanical design, hundreds of specimens and dozens of fuel pins have been irradiation tested in the JOYO and BOR-60 units to investigate irradiation performance. The ODS fuel pins irradiation test in the BOR-60 was executed prior to the JOYO irradiation tests. The BOR-60 irradiation test has continued since 2003 under a collaborative programme between the JAEA and the Research Institute of Atomic Reactors. The peak neutron dose is targeted at 75 dpa and maximum temperature at 973 K. The first irradiation tests under which cladding mid-wall temperatures were 943 and 993 K were already completed without fuel pin failure and a burnup of 50 GW·d/t and a neutron dose of 21 dpa were achieved [12, 13].

To evaluate the irradiation effect up to high neutron dose, the material irradiation test of ODS steel using the core material irradiation rig will be executed in the JOYO facility and the irradiation data to a neutron dose over 200 dpa. The irradiation data concerning the tensile properties of ODS steel cladding tubes were already obtained at temperatures between 673 and 823 K up to a neutron dose of 15 dpa in the JOYO facility. The strength and ductility of irradiated ODS steel claddings were not degraded [14]. The in-pile creep rupture tests using pressurized tube specimens were also carried out by the material testing rig with temperature control (MARICO) in the JOYO facility. As a result of MARICO-2, the irradiation effects on creep rupture strength were negligible up to a neutron dose of 20 dpa and for 4700 h. On the basis of these results, the MSS will be upgraded for the MONJU and JSFR fuels.

5.2. Joining/welding procedures qualification

A further relevant issue is the development and assessment of suitable joining and welding technologies of the selected materials in different geometries (e.g. cladding tubes, heat exchanger). The investigation of various weld techniques (for ODS and F/M steels) to be selected for different applications is a key issue for the applicability of these materials. In GETMAT, the activities envisaged are the welding of claddings and components made of either F/M steels or ODS alloys. Fusion welding technologies such as electron beam and TIG will be investigated for their applicability to F/M steels and F/M–ODS dissimilar welds by defining an improved welding procedure. Welding alternatives to the fusion welding technology, such as electromagnetic pulse, diffusion bonding, friction stir welding and explosive welding, will be investigated as methods of joining ODS alloys. The expected results derive from the assessment of the different weld/join techniques and a ranking of these techniques with respect to the quality of the weld and to their technological applications (e.g. claddings, components).

5.3. Development and qualification of corrosion protection barriers and hard coatings

Experimental results have shown that in Pb/Pb–Bi eutectic, Al_2O_3 scales can prevent the structural materials from becoming heavily oxidized or corroded. Therefore, the surface modification of reference structural materials can mitigate the high demand on the materials in terms of corrosion resistance at high temperature. Activities carried out in the past have confirmed that GESA modified surfaces [15, 16] can resist exposure to the Pb/Pb–Bi eutectic environment for a long time. Therefore, the objective of optimizing the surface alloying procedure to create a ‘smart’ surface layer is of relevance in this field. In particular, the aim is to guarantee a defect free surface layer and a perfect metallic bond at the former interface in order to assure the capability to grow stable protective oxide layers. The development of a method to determine the quality of the GESA modified layer before its use will also be an important and relevant task for licensing purposes.

The smart surface layers will be tested in terms of corrosion, mechanical and fretting resistance. Finally, the behaviour of GESA modified samples in an irradiation field will be assessed through relevant post-irradiation experiments.

Cobalt based alloys, owing to their proven performance, are used for high temperature hard facing applications. However, in consideration of induced radioactivity from ^{60}Co isotopes, nickel based alloys have been chosen for hard facing of PFBR components. As the hard facing is highly susceptible to cracking,

FIG. 9. Hard facing of grid plate assembly of PFBR.

and in order to produce a crack free deposit, special technology has been developed and successfully implemented for hard facing the bottom plate of the grid plate assembly (Fig. 9).

6. MULTISCALE MODELLING AND EXPERIMENTAL VALIDATION

Physical phenomena related to the synergistic effects of irradiation and environment are not supposed to be linear. Incubation times or doses and thermally activated processes may determine the appearance of totally unexpected material responses above a certain dose or temperature. Thus, a safe extrapolation of the behaviour of materials such as high Cr F/M steels to the envisaged in-service conditions must be based on some degree of physical understanding of the basic mechanisms acting on the atomic to the macroscopic levels and on determining their response to the applied environmental, thermal and mechanical loads, while being exposed to neutron irradiation.

Radiation effects on materials are inherently multiscale phenomena in view of the fact that various processes spanning a broad range of time and length scales are involved. The pertinent process includes a wide range of scales from atomic size to a structural component that spans more than 15 orders of magnitude. The timescales also extend from femtosecond to decades. The modelling effort aims at understanding the physical mechanisms forming the basis of the response to, mainly, irradiation of FeCr alloys, as model alloys for high Cr F/M steels. For this purpose, a multiscale modelling approach, including computer simulation tools and an extensive model experiment programme to validate the prediction of codes, is needed. India's efforts involve using ab initio calculations at the atomic level, molecular dynamics to study the evolution of high energy cascades and reactions. The database generated by molecular dynamic simulations and the kinetic Monte Carlo method predict the evolution of local microstructures under irradiation over diffusional lengths and timescales up to seconds. On the basis of the predicted microstructures, dislocation dynamics simulations predict mechanical property changes in the irradiated materials. Validation of these models requires selected parametric experiments under controlled conditions using ion accelerators and available reactors. There are concerted efforts in this direction in Europe (e.g. GETMAT) and elsewhere.

7. FUTURE DIRECTIONS

The high burnup capability of SFR fuels depends significantly on the irradiation performance of their component materials. The JAEA has been developing ODS ferritic steels and a precipitation hardened ferritic steel as the most prospective materials for fuel pin cladding and duct tubes, respectively. Similarly, GETMAT will have a definite impact towards the improvement of knowledge on ODS alloy fabrication, shaping and joining/welding, as well as on their performance in the neutron irradiation field, which is quite limited at present.

Anisotropy in microstructure has been circumvented by subjecting the final clad tubes to austenite–martensite phase transformation in the case of martensitic ODS steels and by recrystallization treatment for ferritic ODS steel clads. While the former one is completely free from anisotropy, ferritic ODS steels still suffer from slight anisotropy in their microstructure. Martensitic 9Cr–ODS steel has higher radiation resistance and formability whereas ferritic 12/13Cr–ODS steel has better corrosion resistance. The corrosion resistance of 9Cr–ODS steel is not good enough in supercritical pressurized water and the Pb–Bi eutectic at high temperatures and there are issues in fuel reprocessing in closed fuel cycles. Thus, for 9Cr–ODS steels, the most critical issue is to improve their corrosion resistance. Currently, material development of martensitic 9Cr–ODS steel and ferritic 12/13Cr–ODS steel are being pursued concurrently. However, there is a need to converge to one of the alloy systems for generating extensive highly expensive irradiation data in a collaborative manner and for easy codification of the steels in standards.

Technology for small scale manufacturing is already established and several hundred ODS steel cladding tubes and dozens of precipitation hardened steel duct tubes have been successfully produced. Development of manufacturing technology for mass production to supply these steels for future SFR fuels needs to be undertaken. Development and incorporation of stringent quality assurance procedures in the manufacturing process using advanced non-destructive evaluation techniques need special emphasis.

Special attention has to be put on the qualification and ranking of promising welding technologies, which are essential for the construction of components, and on corrosion protection methods, e.g. smart coatings, in order to address specifically the protection of the structural materials against corrosion, erosion and possibly friction, thus reducing, if possible, the burden on the structural materials.

The modelling will provide the basis for the physical understanding of the rationalization of the experimental results, which is very much needed in consideration of the fact that the operating conditions of most future reactor concepts cannot be reproduced by any existing facility, so that extrapolation exercises will eventually be needed.

The need for a high breeding ratio and transmutation of long lived actinides necessitate evaluation of metallic fuel and reassessment of core component materials and back end technologies. Advanced non-destructive evaluation techniques for assessment of manufactured components and in-service inspection have been developed, enhancing confidence in the performance of the plant components and systems. The ultimate objective would be to shift from the present domain of materials limited life of components to design basis lifetime performance of components without materials failure. These objectives, complemented with enhanced design inputs, i.e. 60–100 years design life of reactor and 200 000 MW·d/t burnup of fuel, would result in improved economics, safety and reliability of plant performance. Meeting these objectives is the key to realization of the vision of providing fast reactor electricity at an affordable and competitive price.

8. SUMMARY

The fast spectrum reactor employing a closed fuel cycle is an obvious option capable of providing energy security and one which can, potentially, help in addressing the nuclear waste issue. Materials development and materials technologies, particularly the widely used austenitic stainless steels, 9–12Cr F/M steels and ODS steels discussed in this paper, have a deterministic influence on the advancement and success of the fast spectrum reactor programme. Stainless steels are exposed to the challenging environments posed by radiation, temperature, stress and chemicals. Rigorous R&D for alloy development complemented by detailed structure–property evaluation of relevant mechanical and corrosion behaviour data have been possible. These data provide useful inputs for design engineers to ensure reliable and safe operation of the components. Advanced concepts in alloy design and grain boundary engineering have been utilized to enhance the corrosion resistance and mechanical properties of alloys.

Fast spectrum reactor programmes are being steered by giving emphasis to the development of clean steels with low trace and tramp elements, characterization of microstructure and phase stability under irradiation. There is also synergy in codifying mechanical design rules for high temperature structural materials. The rapid development of materials requires a fundamental understanding and a robust predictive capability of radiation damage in materials located in high flux regions. A well-planned synergistic approach for finding robust solutions to materials performance programmes would bring significant benefits, including optimizing cost and time. Significant opportunities exist for the sharing of information on the technology of irradiation testing, advanced methods of microstructure and mechanical properties measurement, safe windows for metal forming and development of a common materials property database system.

Development of a reliable MSS that meets the requirements of the 60–100 year design is another essential issue. The establishment of the physical basis to allow extrapolation of material properties over the long term, which is necessary for the development of the standard, is of prime importance. To cover the inevitable uncertainties associated with such extrapolations, very long term material tests that will accompany the operation of the plant will provide indispensible information. This kind of information would contribute to validating the standard first and subsequently will also serve to optimize the margins involved in the material standard, which has been developed from relatively short term data with ample margins. Monitoring material performance by various advanced technologies would also be beneficial for this purpose. For the next generation of plants, optimization of safety (design) margins associated with material properties would be a key factor to realizing liquid metal FBRs which meet the required levels of safety, reliability and economic competitiveness.

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FUELS AND FUEL CYCLES

(Plenary Session 5)

Chairpersons

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FAST REACTOR FUEL DEVELOPMENT IN JAPAN

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Abstract

The future fast reactor and its fuel cycle system under development in Japan uses oxide fuel with simplified pelletizing fuel fabrication technology as a reference concept. Its driver fuel consists of large diameter annular fuel pellets, oxide dispersion strengthened ferritic steel cladding fuel pins with a ferritic-martensitic steel subassembly wrapper tube and minor-actinide-bearing oxide fuel. The target burnup of the driver fuel is 150 GW·d/t in discharge average, which corresponds to 250 GW·d/t of peak burnup and 250 dpa of peak neutron dose. Fuel developmental efforts, including out-of-pile studies such as material characteristics experimental evaluation and fuel property measurements, various irradiation tests and fuel fabrication technology developments were planned and are in progress. Future fuels will be realized through Joyo irradiation tests and Monju demonstrations. International collaborative efforts are also an important part of such activities.

1. INTRODUCTION

Japan launched the Fast Reactor Cycle Technology Development (FaCT) project in 2006 [1]. The primary concepts of future fast reactors and fuel cycle systems in the FaCT project consist of sodium cooled fast reactors (SFRs), advanced aqueous reprocessing and simplified pelletizing fuel fabrication. In the FaCT project, design studies and R&D on innovative technologies regarding the main concepts are conducted in order to demonstrate and commercialize demonstration and commercial fast reactor cycle facilities by around 2015. This activity will be followed by further developmental efforts to realize the demonstration reactor by around 2025 and the first commercial fast reactor before 2050.

The FaCT project applies advanced oxide fuel concepts as a reference. The current fast reactor fuel development programme is concentrating on the reference oxide fuel of the FaCT project.

2. FUELS AND CORE MATERIALS USED AT JOYO AND MONJU

Japan has constructed two fast reactors, Joyo [2] and Monju [3]. Table 1 shows their driver fuel specifications. Fuel pin diameters are 5.5 mm and 6.5 mm

TABLE 1. DRIVER FUELS USED AT JOYO AND MONJU

			Joyo (MK-III)	Monju
Sub-assembly				
	Overall length	(m)	2.97	4.2
	Distance between flat	(mm)	78.5	110.6
	Flow rate range	(kg/s)	6.8 to 8.5	14 to 21
Pin				
	Overall length	(mm)	1533	2813
	Fuel column length	(mm)	500	930
	Diameter (Inner/outer)	(mm)	4.8/5.5	5.56/6.5
	Spacer		wire	wire
	Triangular pitch	(mm)	6.47	7.87
	Number of pins		127	169
Pellet				
	Type		solid	solid
	Diameter x Height	(mm)	4.63 x 9	5.4 x 8
	Smeared density	(% TD)	87	80

for Joyo and Monju, respectively. Fuel pellets are mixed oxide solid pellets. Cladding materials of the Joyo driver fuel are PNC316 [4], which is modified type 316 steel and PNC1520 [5, 6], which is an advanced austenitic steel with 15% Cr and 20% Ni. PNC316 is also a current Monju driver fuel cladding. The Japan Atomic Energy Agency (JAEA) has accumulated significant experience with mixed oxide fuel fabrication for Joyo and Monju, and significant irradiation experience for Joyo, including irradiation tests.

Joyo has the extended capability to conduct a range of irradiation tests by applying various irradiation rigs, including material test, fuel pin test, fuel pin bundle test and off-normal condition test such as fuel power-to-melt test and has been utilized as an excellent irradiation tool for fast reactor fuel development.

Figures 1 and 2 show irradiation conditions achieved in Joyo irradiation rigs. Figure 1 shows burnups and linear heat rates achieved by fuel irradiation tests. Figure 2 shows fast neutron fluences and irradiation temperatures achieved by material irradiation tests. Achieved irradiation conditions are beyond those of the Monju driver fuel. Fuel irradiation tests include high linear power tests to investigate the fuel centre line melting limits and large diameter annular pellet fuel pins as a typical advanced concept. Core materials in material irradiation tests include oxide dispersion strengthened (ODS) ferritic steel [7], which is a reference cladding material for future high burnup fuel pins, high strength ferritic-martensitic steel (PNC-FMS) [6], which is a reference subassembly

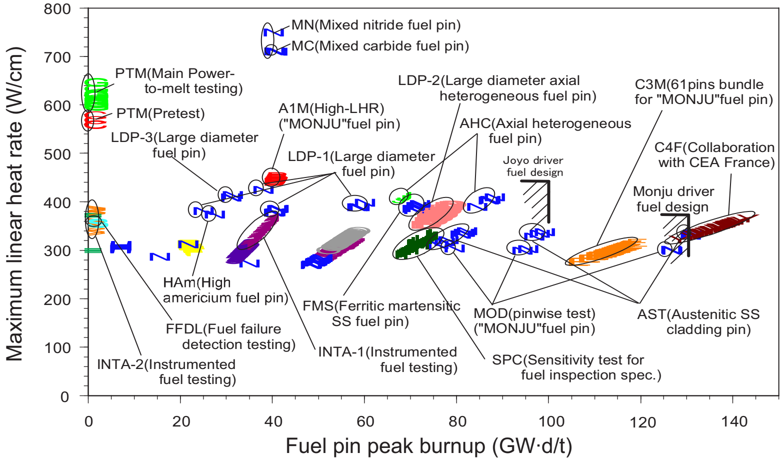

FIG. 1. Results of fuel irradiation tests at Joyo.

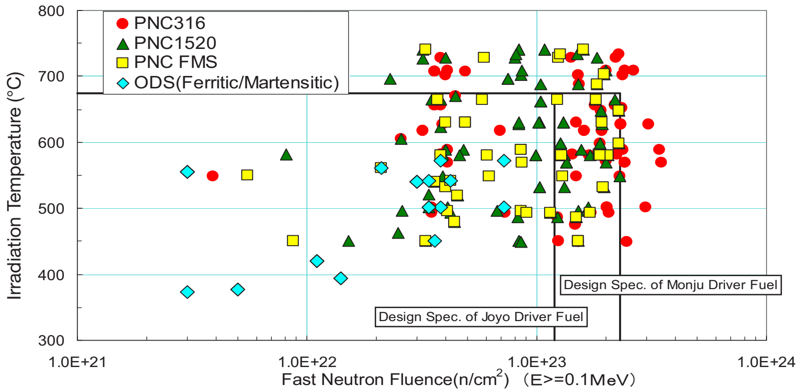

FIG. 2. Results of material irradiation tests at Joyo.

wrapper tube material for future high burnup fuel subassemblies, as well as PNC1520 and PNC316.

Monju is being prepared to be utilized for fuel subassembly demonstration of future fast reactor fuels. Joyo and Monju will fill a role in fast reactor development and will progress fuel development for the FaCT project and future commercial reactors.

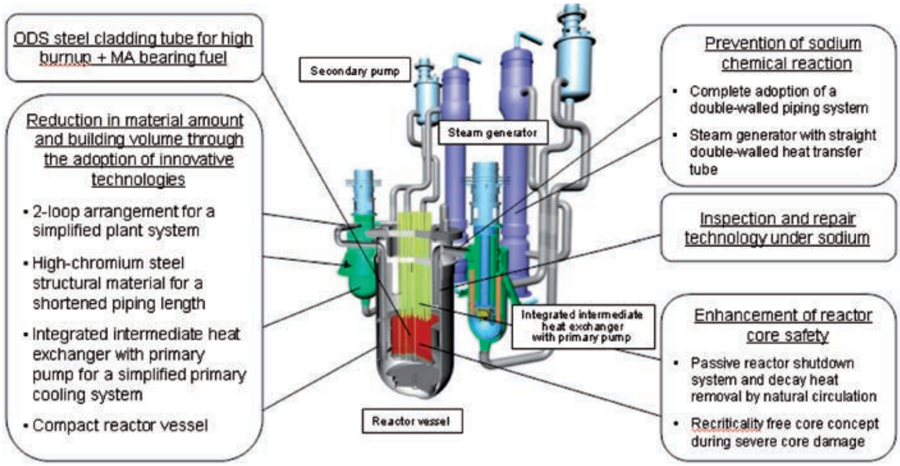

FIG. 3. Main features of the JSFR.

3. REFERENCE CONCEPTS OF THE FUTURE FAST REACTOR AND ITS FUEL CYCLE [1, 8]

Figure 3 shows a reference concept of an SFR system, the Japan sodium cooled fast reactor (JSFR). The JSFR concept includes innovative technologies such as a tow loop system in the primary circuit and a passive shutdown system of the reactor. For the core and fuel, a high burnup core with ODS steel cladding and minor-actinide-bearing (MA-bearing) oxide fuel are proposed.

Figure 4 shows the reference concepts of reprocessing and fuel fabrication. Major features are the homogeneous recycling of MAs and a simplified pelletizing process. The simplified pelletizing process is intended to reduce the fabrication process steps by supplying starting oxide fuel powder with adjusted enrichment and by adopting a pellet pressing method of using binderless granulated oxide. Its advantage of a reduced number of pellet fabrication steps is indicated in Fig. 5. This process is expected to reduce the cost and waste of fabrication and is applicable both to MA-bearing oxide fuel and to (U,Pu) oxide fuel without intended MAs.

The JSFR core and fuel are designed to achieve high burnup and a high core outlet temperature, and be capable of operation with homogeneously recycled MA-bearing fuel, not only for fast reactor recycle, but also for a fuel supply from LWR spent fuel. Its reference fuel is oxide fuel and the alternative is metal fuel. In the case of oxide fuel, an MA content of up to 5% in heavy metal is considered.

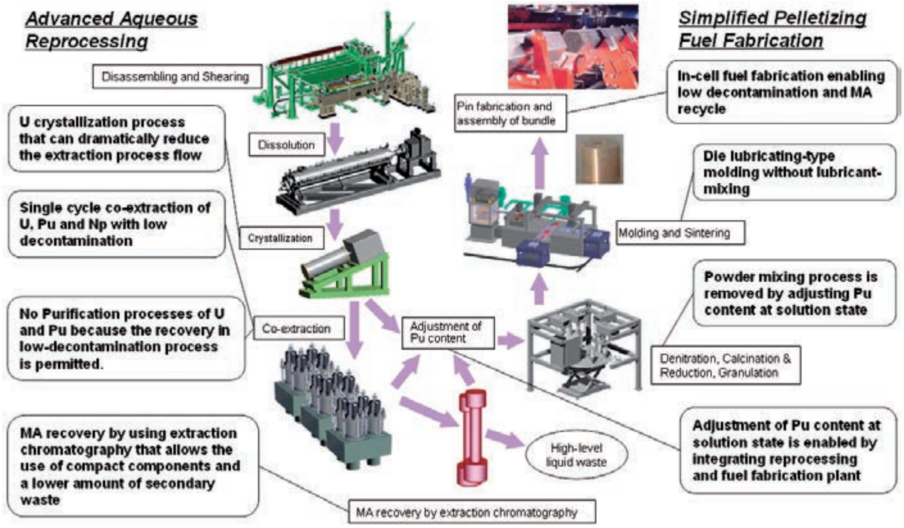

FIG. 4. Main features of the advanced fuel cycle system of the FaCT project.

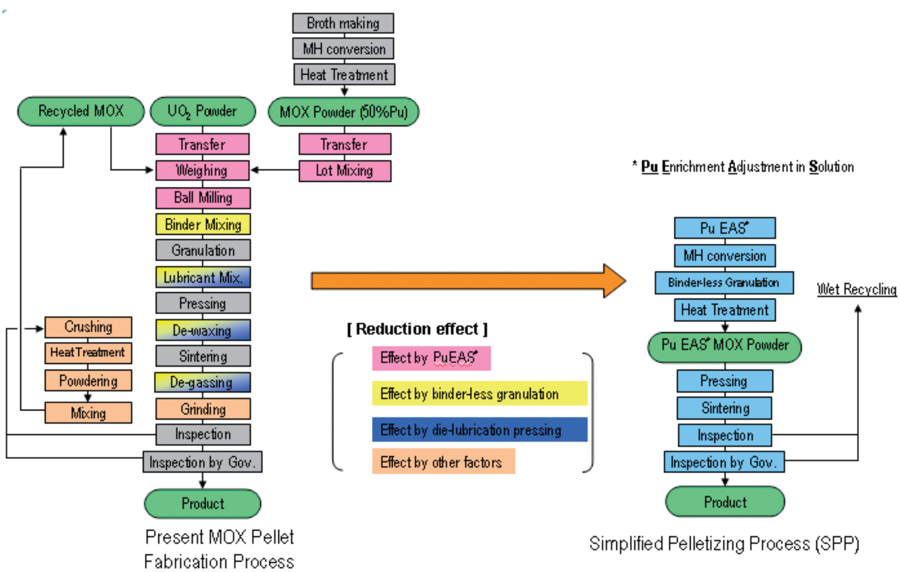

FIG. 5. Flow chart of the simplified pelletizing process.

Core and Fuel Specifications

Items	Breeding Core	Break Even Core
Nominal full power (MWe/MWt)	1,500/3,570	←
Coolant temperature [outlet/inlet] (°C)	550/395	←
Primary coolant flow (kg/s)	18,200	←
Core height (cm)	100	←
Axial blanket thickness [upper/lower] (cm)	20/20	15/20
Number of fuel assembly [core/radial blanket]	562/96	562/ —
Envelope diameter of radial shielding (m)	6.8	←
Fuel cladding outer diameter (mm)	10.4	←
Fuel cladding thickness (mm)	0.71	←
Number of fuel pin per assembly	255	←
Wrapper tube outer flat-flat width (mm)	201.6	←
Wrapper tube thickness (mm)	5.0	←

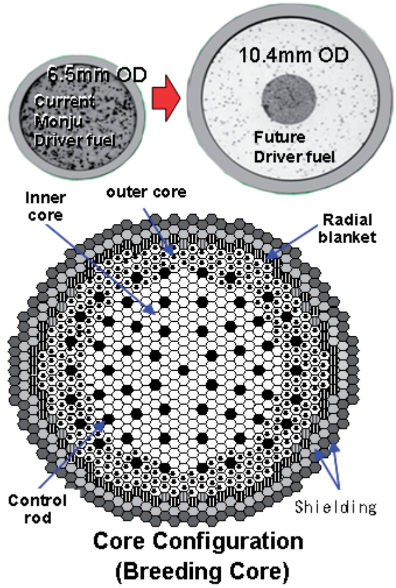

FIG. 6. Large scale JSFR core and fuel concept.

Figure 6 shows a reference concept of JSFR core and fuel. The core outlet temperature is 550°C. The core fuel column length is 100 cm. The fuel pin has a large diameter, about 10 mm, and a high density annular pellet fuel to have a low smeared density of 82% theoretical density to achieve high burnup with high density pellets. This is a change from the current Monju driver fuel pin, which is 6.5 mm in diameter and uses low density solid fuel. Such a fuel pin design gives the advantage of a high fuel volume fraction in the core to realize superior core neutronic characteristics, a low fuel smeared density to accommodate fuel swelling at high burnup and fuel fabrication economics consistent with low fuel smeared density design. A low oxygen/metal fuel is also used to reduce the cladding inner surface corrosion at high burnup. In aiming to achieve high burnup, 150 GW·d/t of the average burnup, which corresponds to 250 GW·d/t of peak burnup and 250 dpa of a fast neutron dose, an ODS ferritic steel cladding and a PNC-FMS subassembly duct were selected as reference core materials. Using an ODS cladding tube with high strength at high temperatures, a design with 700°C maximum cladding temperature is feasible.

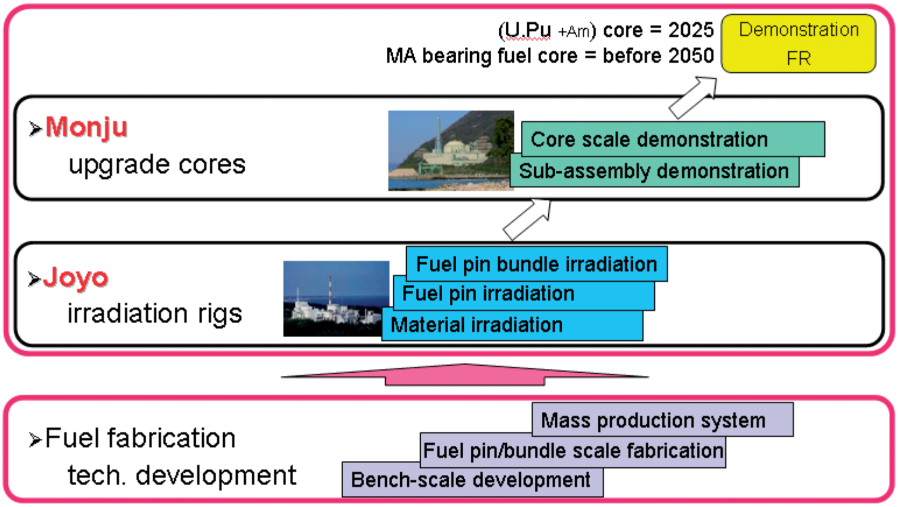

FIG. 7. Basic scheme of fuel development for the FaCT project.

4. FUEL DEVELOPMENT

Fuel development activities are in progress to realize the JSFR driver fuel of the FaCT project. Figure 7 shows a basic scheme of fuel development. Using Joyo irradiation rigs, material irradiation tests, fuel pin irradiation tests and fuel pin bundle irradiation tests will be performed. On the basis of these irradiation experiences and on test data, subassembly demonstration irradiation and core scale demonstration will be done in the Monju upgrade cores. These are leading irradiation tests to realize the demonstration reactor cores, which are expected to start in approximately 2025. Fuel fabrication technology development will supply the fuels needed for these irradiations.

Major irradiation tests to be performed in Joyo and in other reactors are:

(a) ODS irradiation (material, fuel pin, fuel pin bundle);
(b) PNC-FMS irradiation (material, fuel pin, subassembly duct);
(c) Large diameter fuel pin;
(d) Simplified process fuel pellets;
(e) Annular fuel power-to-melt;
(f) Irradiated fuel power-to-melt;
(g) MA-bearing oxide fuel (Am+Np-bearing, Am+Np+Cm-bearing);
(h) Transient tests (reactor tests and hot cell tests);
(i) Burnup extension of current fuels.

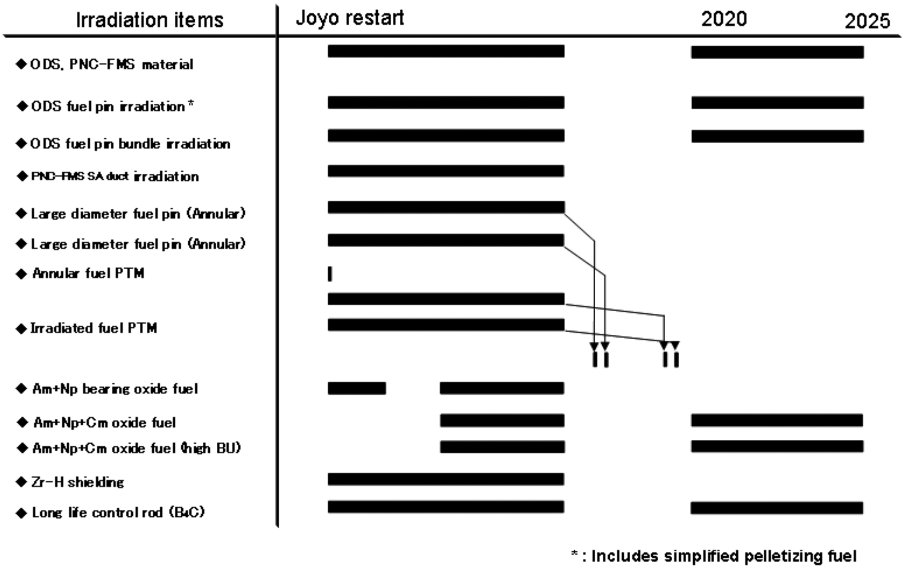

FIG. 8. Joyo irradiation tests for the FaCT project.

Joyo irradiation tests are indicated in Fig. 8. Most of the results will be obtained before 2020 and some of them will be continued until around 2025.

The fuel property study is a significant issue for MA-bearing oxide fuel. The JAEA has been investigating MA-bearing fuel properties through the out-of-pile experimental studies and the analytical studies [9]. Major properties, such as fuel melting point, thermal diffusivity and specific heat, have been studied experimentally. Current results show the limited contribution of MAs on the properties of MA-bearing, homogeneously recycled oxide fuels.

Fuel fabrication technology is also a key issue of future fast reactor fuel. The simplified pelletizing process under development aims at reducing the pellet fabrication processes owing to fewer oxide fuel powder treatment processes and fewer organic additives [10]. Its key technologies are homogeneous oxide powder supplied by the microwave conversion process of the U–Pu solution, binderless granulation and die lubrication pressing. The microwave conversion process has already been established and development of the other two technologies is progressing well. The irradiation behaviour of fuel pellets made from microwave conversion powder was also investigated at a preliminary stage.

4.1. MA-bearing oxide fuel irradiation test

Some of the irradiation tests have already started. The MA-bearing oxide fuel irradiation test is one such irradiation test, where Am- and Np-bearing fuel

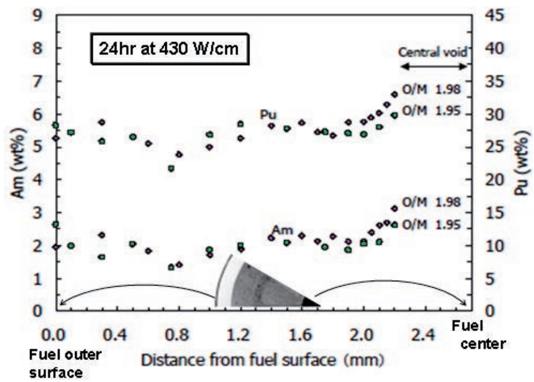

FIG. 9. Am and Pu redistribution of Am+Np-bearing MOX fuel in Joyo [11].

pins were irradiated in Joyo for up to 10 min and 24 h at maximum power [11]. A short term irradiation test of the MA-bearing oxide fuel supplied important irradiation data, such as early-in-life fuel restructuring of the MA-bearing fuel and the MA redistribution, as well as Pu redistribution. It is very important to know the MA redistribution behaviour at the beginning of life, since it may limit the maximum linear heat rate of MA-bearing fuel due to the melting limit of the licensing design evaluation. Irradiation test results of MA-bearing fuel in Joyo show that the fuel restructuring behaviour is identical with (U,Pu) fuel without MAs, and that Pu and Am redistribution is limited, especially in low oxide/metal fuel, as shown in Fig. 9. This was qualitatively expected and will be applied to quantitative analysis of future driver fuel designs. The irradiation test will continue to supply further irradiation data, leading to subassembly demonstration irradiations in Monju.

4.2. ODS cladding irradiation test

The ODS cladding irradiation tests in the BOR-60 for the fuel pin irradiation test [7] and in Joyo for the material irradiation test are in progress. Interim results of the BOR-60 irradiation test showed little cladding inner surface corrosion at the cladding maximum temperature of over 700°C. The result of 100 GW·d/t burnup in BOR-60 also shows little corrosion, as indicated in Fig. 10. A higher burnup irradiation test is under preparation for Joyo irradiation and under discussion for the BOR-60 irradiation test.

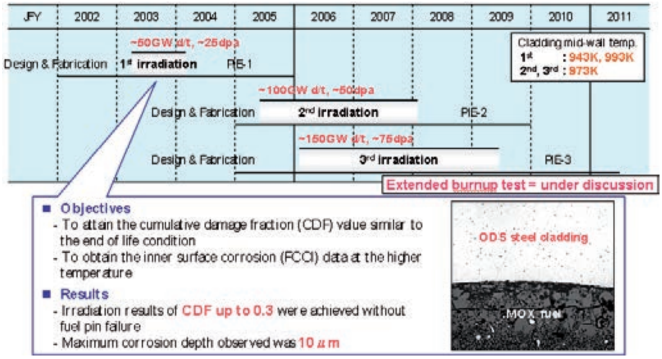

FIG. 10. ODS cladding fuel pin irradiation tests [7].

4.3. Global Actinide Cycle International Demonstration (GACID)

International collaboration is indispensible to the promotion of fast reactor fuel development. The Generation IV International Forum (GIF) is one of the relevant frameworks for such collaboration. One such example is GACID, a collaboration between the CEA (France), the USDOE (United States of America) and the JAEA under the GIF Sodium-cooled Fast Reactors (SFR) System Arrangement [12]. The GACID project aims to demonstrate the MA transmutation capability and MA-bearing fuel integrity in a fast reactor core, using Joyo and Monju. The project consists of three steps comprising a series of irradiation tests in Joyo and Monju:

(1) Step 1: Precedent limited MA-bearing fuel preparatory irradiation test
This test assumes ^{237}Np and ^{241}Am are the only MAs. Moreover, only a single pin scale irradiation test in Monju is planned. Therefore, this test is expected to be implemented at the earliest stage of the project.

(2) Step 2: Pin scale Cm-bearing fuel irradiation test
A full range of MA compositions is assumed for this test. Not only Np and Am but also Cm will be contained in the test fuel, although the test will be conducted on a pin scale. A precedent irradiation test in Joyo is being planned for the Monju irradiation licensing.

(3) Step 3: Bundle scale MA-bearing fuel irradiation demonstration
After completing the above-mentioned two steps of the precedent irradiation tests, the final goal, the bundle scale full range MA-bearing fuel irradiation demonstration, will be performed in Monju. This technical

demonstration will be done in a reasonable timeframe and the whole project is to be conducted over a period of 20 years.

The project is in progress and the following activities are under way in each participating organization:

(a) MA raw material preparation and shipping;
(b) MA-bearing MOX fuel pellet sintering;
(c) Material property measurement;
(d) Precedent Joyo irradiations and post-irradiation examinations;
(e) Licensing at Monju and Joyo;
(f) Preliminary programme planning for bundle scale irradiation demonstration.

The GACID project is a good example of international collaboration that accelerates the development programme and provides efficient results.

4.4. Metallic fuel irradiation tests

The U–Pu–Zr metal fuel is an alternative fuel for the JSFR. It is well understood that the U–Pu–Zr base metal fuel developed by Argonne National Laboratory has, as its major drawback, the steel cladding temperature limit due to fuel–cladding compatibility factors.

In collaboration with the Central Research Institute of Electric Power Industry, irradiation tests to investigate the fuel–cladding compatibility issues at high temperatures will start with six metallic fuel pins with a PNC-FMS cladding tube, which is high strength cladding [13]. The target burnups are 3 at.%, 8 at.% and over 10 at.%. The fuel smeared density and peak cladding temperature are parameters. The ODS cladding tubes are expected to be applied to the extended phase of this experiment. The first result will be obtained after a few irradiation cycles at Joyo, which will be followed by long term steady state irradiation to obtain high burnup data. The irradiation test results will show the feasibility of attaining a maximum cladding temperature of 650°C without a metallic fuel–cladding compatibility problem arising.

5. CONCLUSIONS

The JAEA is conducting fuel development activities for future fast reactors as a part of the FaCT project in Japan. Developmental efforts include irradiation tests, fuel fabrication technology development and out-of-pile studies, such as fuel property investigations. Future fuels will be realized through irradiation tests at Joyo and demonstrations at Monju. International collaborative efforts are also an important part of such activities.

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FUELS FOR ADVANCED SODIUM COOLED FAST REACTORS IN THE RUSSIAN FEDERATION: STATE OF THE ART AND PROSPECTS

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Abstract

Different fuels (i.e. PuO_2, UO_2 (pellet, vibro), $UPuO_2$ (pellet, vibro), UC, UN, UPuC, UPuN, oxide, nitride and carbide inert matrix fuels, alloyed and non-alloyed metallic fuels) have been studied in BN reactors. Recently, experiments with UPuN, MgO and ZrN based fuels have been completed in the BOR-60 reactor. The paper presents an overview of the principal results of fuel investigations. The problems of reliable fuel performance in the BN-800 and BN-K reactors are discussed.

1. INTRODUCTION

The development of a closed uranium–plutonium fuel cycle with step by step implementation of fast reactors, to provide the required fissile material breeding, is one of the advanced directions taken by nuclear power development in the Russian Federation. The advanced fast reactor with a closed fuel cycle permits realization of the following requirements:

(a) Prevention of heavy reactor accidents;
(b) Prevention of heavy accidents on fuel cycle facilities;
(c) Provision of low waste recycling of nuclear fuel;
(d) Compliance with non-proliferation requirements;
(e) Competitive electricity production;
(f) Nuclear fuel breeding and effective use of fuel resources.

The above listed problems may be solved by the step by step implementation into the Russian nuclear power structure of nuclear units with advanced BN-type reactors of major power (1000–1800 MW(e)). This nuclear technology has passed the stage of semi-industrial mastering and has good enough potential for the development of commercial BN reactors, i.e. high parameters of safety, effective use of fuel resources and competitive with alternative energy sources. However, the optimum method of BN technology realization (reactor design parameters, fuel type, fuel cycle type, optimum parameters of fuel consumption, etc.) is not currently defined and additional consideration and investigations are required.

A comprehensive and consistent programme of sodium cooled fast reactor (SFR) development has been implemented in the Russian Federation: BR-1 (1954) →BR-2 (1956) →BR-5 (1959) →BR-10 (1973) →BOR-60 (1969) → BN-350 (1973) →BN-600 (1980) →BN-800 (under construction) →BN-1800, BN-1200 (under development). In the BR-5, BR-10, BOR-60, BN-350 and BN-600 reactors, the different fuel types (PuO_2, UC, UN, UPuN, UO_2, $UPuO_2$, alloyed and non-alloyed metallic fuels, inert matrix fuels) have been irradiated and investigated. Currently, the investigations are carried out in the BOR-60 and BN-600 reactors.

2. FUELS FOR ADVANCED REACTOR CORES

2.1. BN-800 reactor

Currently, the construction of the fourth power unit of the Beloyarskaya NPP with the BN-800 reactor is quite important in order to increase competitiveness and safety, create the components of the fast reactor fuel cycle and decrease its environmental impact. The BN-800 core design employing MOX fuel has been developed. The decision was made to use vipac MOX fuel as a standard BN-800 fuel. The maximum fuel burnup is about 10 at.% and the maximum dose is 90 dpa. The fuel assembly wrapper is made of EP-450 (13Cr–2Mo–Nb–P–B) ferritic-martensitic steel and the pin cladding of ChS-68cw (16Cr–15Ni) austenitic steel. In the framework of the national programme, work on the improvement of ChS-68 irradiation stability has been carried out; the first stage of BN-600 irradiation of experimental fuel pins with cladding made of EK-164cw (16Cr–19Ni) has been successfully completed. The aim of these investigations is to limit the maximum dose to 110 dpa for the BN-800 core.

The specific feature of the core design is its reversibility, allowing use of a nitride core. From domestic and foreign experience, the requirements for nitride fuel can be defined in order to be reliable under irradiation in fast sodium reactor cores. First of all is the increased fuel porosity value (up to 15–20%). It fails to meet the requirement for the BN-800 core design developed for the MOX fuel. The calculations show that the allowable value of nitride burnup for a helium bonded BN-800 pin does not exceed 9 at.%.

2.2. BN-K (commercial) reactors: BN-1800 and BN-1200

The basic BN-K core design requirements are:

(a) Closed fuel cycle with a minimum amount of radwaste;
(b) Minimum level of core breeding (~1) that permits a decrease in the reactivity excess of $<\beta_{eff}$;
(c) Core design should provide for breeding, with the breeding ratio up to 1.45 for dense fuel types;
(d) High burnup and increased operation cycle in order to improve the economy.

Currently, MOX fuel is the reference fuel for the commercial BN-K reactor. The increase of core breeding for the MOX fuel core is provided by the higher fuel volume fraction and the increased fuel smeared density of up to 9.2 g/cm^2. However, better physics parameters are provided by a mixed nitride core. The nitride core is more compact, has higher core breeding and breeding ratio, permits a decrease in the excess reactivity, not only per burnup but also per temperature power effects. However, additional studies are required in order to demonstrate the possibility of high nitride burnups and nitride core safety. Therefore, nitride fuel is an additional option.

In order to provide high fuel burnup, the ferritic-martensitic steels are considered as a cladding material. Their oxide dispersion strengthened (ODS) modifications are under investigation as well, aiming to increase high temperature stability. The domestic investigations of ODS steels were stopped more than 15 years ago and restarted only several years ago. The ODS steels should provide an increase in burnup value up to ≥20 at.%, as well as temperature parameters and reactor efficiency.

The EP-450 steel is planned to be used as a wrapper material. There is some experience in the use of this steel as a cladding material in the BOR-60, BN-350 and BN-600 reactors. The experimental fuel pins with EP-450 cladding and vibropacked MOX fuel have been successfully tested in the BOR-60 reactor up to 142 dpa. However, the low level of EP-450 high temperature strength, compared with the ChS-68cw and EK-164cw austenitic steels, restricts its application as a cladding material in the BN-600 and BN-800 reactors. In order to provide the possibility of using ferritic-martensitic steels in the BN-K reactors, it is planned to decrease the maximum cladding temperature to 660–670°C (instead of 700°C as in the BN-600 and BN-800). This is achieved by means of a sodium temperature rise decreasing in the core (up to 140°C). Besides the investigations of complex alloyed EK-181, ChS-191 steels with higher temperature strength than EP-450 are under development. These steels are characterized by additional alloying of C, N, W, Ta and some decrease of the Cr content [1].

With the orientation on the improved ferritic-martensitic steels, the conceptual design of the BN-1800 reactor with a pellet MOX fuel core (17 at.% of maximum burnup (stage I) and 20 at.% (stage II)) and the technical proposal for the BN-1800 reactor with a nitride core were developed (maximum burnup: 13 at.%, maximum dose: 160 dpa) in 2003. The transition to mixed nitride may be done without changing the oxide core and fuel assembly designs. The smeared density of nitride with natural nitrogen is ~80%.

Currently, R&D work is under way on the development of a commercial fast sodium BN-1200 reactor with the maximum use of already time tested and scientifically based technical decisions realized on the BN-600 and BN-800 designs. The core concept and burnup level are the same for the BN-1800. The possibility of minor actinide utilization is also assumed. Two options for minor actinide utilization are under consideration: (i) a homogeneous type with small minor actinide additions to standard fuel and (ii) a heterogeneous type (in special targets).

In order to increase the breeding ratio, the heterogeneous cores are considered as well, in which the depleted alloyed or unalloyed metals are used. One of the oxide–metal core models with axial heterogeneity is compatible with the BN-1200 core under design and has a high breeding ratio (~1.4) and almost zero change of reactivity with burnup.

3. PRINCIPAL RESULTS OF FUELS STUDY

3.1. MOX fuel

Two MOX fabrication techniques (vibropacking and pellets) have been developed in the Russian Federation. The MOX pellet fabrication methods were developed at VNIINM, Moscow. Two methods were used for fuel powder preparation: (i) mechanical mixing of initial PuO_2 and UO_2 (the MMO method) and (ii) mixed co-precipitation of U and Pu dioxides (the GRANAT method).

The mechanical mixing method consists of single stage milling–blending of individual powder oxides with subsequent granulation, pressing and sintering of the prepared blend. The major problem of the mechanical mixing method is in achieving the blend homogeneity for meeting the requirements imposed on fuel to ensure uniformity of plutonium content in the volume of a fuel pellet. In addition, easy reprocessing of spent fuel should also be ensured. A ball mill is widely used in global practice as a blender to meet the requirements on blend homogeneity. The vortex blender is used for this purpose in the Russian process of pelletized MOX fuel fabrication. When introduced in the process, the vortex blending undertaken in a rotating electromagnetic field made it possible to improve the fuel homogeneity substantially. The uniformity of plutonium distribution in a fuel pellet fabricated using this method is comparable to that obtained for the chemically co-precipitated fuel. When mechanically processed in the mixer (blender), the powder takes on special properties, contributing to more active sintering of a compressed pellet in the course of subsequent heat treatment. The process of homogeneous powder mixture fabrication has received the name vortex milling process (VR process), derived from the name of the device used. The VR process is universal and permits the fabrication of both oxides and nitrides of the required quality. The process of formation of the homogeneous oxide powder blend is illustrated in Fig. 1.

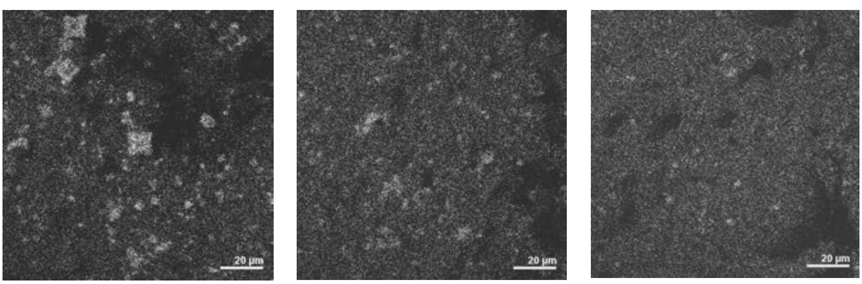

FIG. 1. Milling and blending of powders in AVS-150 (photos of the blend samples in characteristic X rays. The milling–blending time increases from left to right) [2].

The fabrication technique of vibropacking MOX fuel was developed at RIAR, Dimitrovgrad. For the last 30 years, RIAR has carried out investigations of the closed fuel cycle, based on the pyrochemical processes of spent fuel reprocessing and receiving the granulate for the fuel fabrication using the vibropacking method. The pyrochemical reprocessing and vibropacking fabrication technique permit the realization of the remote–automatic type of granulate and fuel pin fabrication. The required quality of granulate is achieved by the combined electrocrystallization of UO_2–PuO_2 from molten alkali metal chlorides. The technology of vibropack MOX fuel fabrication by mixing mechanical oxides has also been developed. The method provides the predetermined Pu distribution and minor actinide inclusion [3].

The BOR-60 and BN-600 fuel designs, as well as parameters of the fabrication route and monitoring under the remote–automatic condition, have been optimized on the basis of fuel testing in the SM, BOR-60 and BN-350 reactors. The fuel column is a mechanical mixture of $(U,Pu)O_2$ granulate and U powder, used as ‘getter’, which is introduced at the mixing stage before the fuel filling. The introduction of a getter for fuel:oxygen ratio adjustment and removal of the influence of technological impurities has provided an option for removal of the fuel–cladding chemical interaction.

The vibropacked co-precipitated MOX fuel is the BOR-60 driver fuel with maximum burnup of 15 at.%. The maximum burnup of 30 at.% is achieved for BOR-60’s several experimental assemblies. The experimental fuel assemblies with pellet and vibropacked MOX fuel have been irradiated at the BN-600 reactor. The maximum burnup of pellet MOX fuel is ~12 at.% and 10.5 at.% for vibropacked MOX fuel.

3.2. Nitride

Domestic experience with regard to irradiation behaviour of nitrides covers the following:

(a) Two loadings of the BR-10 reactor with UN (660 and 590 He-bonded fuel pins);
(b) The BOR-60 He-bonded experimental pins with: (i) UN, maximum burnup of more than 8 at.%, and maximum fuel temperature of 1775 K; (ii) UPuN, maximum burnups of 4 at.% and 9 at.%, maximum fuel temperatures of 2475 K and 1750 K, respectively; (iii) UPuN with increased Pu content (45% and 60%) within the framework of the joint Russian–French BORA–BORA programme, maximum burnups of 9.3 at.% and 12 at.%.

Eighteen years of operation of the BR-10 reactor with UN fuel has shown good performance with up to 8 at.% burnup. More than 99% of fuel pins have reached the design value of burnup (8 at.%) without failure. For the second core (design burnup 8.8 at.%), the number of fuel failures has increased, mainly at burnup values of more than 8 at.%. Twenty-four cases of fuel failure were detected in the second core. The analysis of the results of the researchers' calculations of nitride pin performance shows that the most probable reason for fuel failure is fuel–cladding mechanical interaction. The post-irradiation of six fuel assemblies of reactor loading no. 4 have been carried out. Data on swelling and gas release from nitrides with respect to burnup have been received.

The BORA–BORA programme included three phases: (i) fuel fabrication, (ii) fuel irradiation in the BOR-60 reactor and (iii) post-irradiation examination. The following fuels have been studied [4]:

(a) $UPu_{0.45}O_2$ (4 fuel pins with pellets and 4 fuel pins with vibropack fuel);
(b) Nitride pellets (2 fuel pins with $UPu_{0.45}N$ and 2 fuel pins with $UPu_{0.6}N$);
(c) Inert matrix pellets (2 fuel pins with 40%PuN + 60%ZrN and 2 fuel pins with 40%PuO_2 + 60%MgO).

The irradiation of fuel pins incorporated in two dismountable fuel assemblies started in August 2000. One fuel assembly has 4 pins with MOX pellets, 4 with MOX vibro, and another fuel assembly has 4 nitride and 4 inert matrix fuel pins. In November 2002, two fuel assemblies were unloaded for intermediate post-irradiation examinations at burnups within 5.4–11.3 at.%, depending on the fuel type (first irradiation stage). After the intermediate examination, parts of the fuel pins were discharged for the destructive post-irradiation examination, while the others were taken back for further irradiation beginning on 2 December 2003. The irradiation was completed in May 2005. The last post-irradiation examinations of the $UPu_{0.45}N$ and $UPu_{0.6}N$ fuels were completed in 2009 (Table 1).

Non-destructive and destructive post-irradiation examinations of $UPu_{0.45}N$ and $UPu_{0.6}N$ (Fig.2) have shown that:

(a) The fuel density measurements made at the central axial section have shown that the average swelling rate of $UPu_{0.45}N$ is 0.48–0.68 ± 0.04%/1 at.%, and for $UPu_{0.6}N$ it is 0.64–1.11 ± 0.04%/1 at.%.
(b) The fission gas release under cladding is 19% ($UPu_{0.45}N$) and 19.3% ($UPu_{0.6}N$).
(c) The structure of both fuel types is characterized by intragranular and intergranular porosities. The higher part of $UPu_{0.45}N$ open intergranular porosity promotes better gas release and provides a lower gas swelling rate.

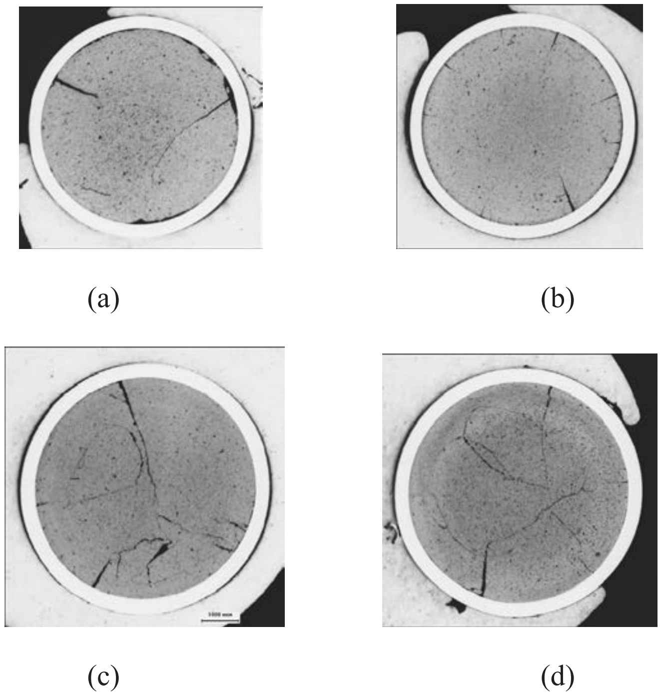

FIG. 2. (a) $UPu_{0.45}N$ — core top, (b) core middle, (c) $UPu_{0.6}N$ — core top, (d) core middle.

(d) The maximum cladding corrosion depth of 15 μm was detected at the upper core section of the $UPu_{0.6}N$ pin.
(e) No nitride dissociation was detected on microstructure or on nitrogen content changes.

The positive results of irradiation tests of high purity mixed nitride at the BOR-60 reactor at up to 12.1 at.% at a maximum linear rating of 54.5 kW/m may be explained by the high initial homogeneity of the Pu distribution, the low oxygen and carbon contents (<0.15 wt% and <0.1 wt%, respectively), the uniform porosity distribution and the combination of intragrain and along grain border pores. The performance of the nitride fuel pin as well as the oxide is limited by the irradiation stability of the cladding steel and also by the fuel–cladding mechanical interaction (BR-10 reactor experience). The positive BORA-BORA results confirm the possibility of providing at least 12 at.% burnup for the He-bonded pins at the initial fuel porosity increase. The BORA–BORA nitrides' porosity is 15%.

3.3. Metal fuel

Two techniques for the alloyed metal fuel fabrication are under study in the Russian Federation: hot extrusion and injection casting. Through the different programmes, including collaboration with KAERI, the BR-10 and BOR-60 fuel pins with U–Zr and U–Pu–Zr have been fabricated. Two full-scale fuel assemblies with U–Pu–Zr have been irradiated at the BOR-60 reactor up to 10 at.%.

As the result of many years' investigations carried out by RIAR, Dimitrovgrad, the basic principles of radiation growth, swelling, gas release and corrosion behaviour of unalloyed U–Pu, with and without protective layers, have been determined. The blanket, absorber and core fuel pins with U and U–Pu have been fabricated and irradiated at the BOR-60 and BN-350 reactors. The test results have demonstrated the performance of He-bonded pins with metal fuel of high smeared density (≥12.5 g/cm^3) [5]. However, the use of metal fuel in BN-type reactors is restricted by the necessarily high sodium temperature which accordingly determines reactor efficiency.

3.4. Inert matrix fuel

Transmutation and incineration are innovative options in the management and disposal of fission products and actinides. In order to improve the efficiency of these processes, materials inert to neutron activation are being considered as alternatives to UO_2 as a support material, as the latter generates actinides during irradiation.

In the framework of the BORA–BORA programme, the irradiation and post-irradiation examinations of two pins with (Pu,Zr)N and two pins with PuO_2+MgO in the BOR-60 reactor have been completed [6]. All pins are intact. The irradiation parameters of the experimental pins are given in Table 2.

TABLE 2. IRRADIATION PARAMETERS OF BORA–BORA INERT MATRIX FUEL PINS [6]

Fuel	Pu content (wt%)	Density (% theoretical)	Maximum burnup (at.%)	Maximum linear rating (kW/m)
PuO_2+MgO	35.8	88–91	11.1 (I stage) 18.8 (II stage)	10.0
(Pu,Zr)N	37.5	83–84	11.3 (I stage) 19.2 (II stage)	20.7

The irradiated 40%PuN–60%ZrN fuel has a two-phase structure consisting of solid solutions based on zirconium and plutonium nitrides of differing proportions and fission product contents. The hydrostatic weighing showed the swelling rate to be less than 0.1%/1 at.%. The gas release is about 1%. The maximum cladding corrosion depth is about 15 μm.

The irradiated 40%PuO_2–60%MgO fuel has a two-phase structure consisting of plutonium and magnesium dioxides with fission products in solid solution. The average value of the swelling rate measured by hydrostatic weighing is about 0.5%/1 at.%. The gas release is 9%. The maximum cladding corrosion depth is less than 10 μm.

In the framework of STC Project #2680 MATINE – Study of Minor Actinide Transmutation in Nitrides: Modelling and Measurements of Out-of-Pile Properties, (Pu,Zr)N samples with a ZrN content of 60 mol% of 85% and 93% theoretical densities meeting specifications have been fabricated by direct nitridation of metals. Powder mixing has been done by a patented electro-vortex blending method (rotating ferromagnetic needles); solid solution was achieved. Experimental studies of the thermophysical properties of (Pu,Zr)N laboratory samples — thermal creep, thermal conductivity, high temperature stability, thermal expansion — have been carried out [4].

Temperature dependence of sample thermal conductivity over a temperature range of 500–1600°C has been studied by a laser flash method under vacuum. Certain discrepancies between the thermal conductivity of (Pu,Zr)N investigated in this research and respective foreign data have been noticed. This can be explained by different physicochemical and structural properties of the samples resulting from differences in sample fabrication processes.

High temperature creep has been studied by the method of uniaxial compression at fixed temperatures under very high purity argon. The experimental data at a steady state phase, depending on the temperature, are satisfactorily described by the Arrhenius equation. The creep rate increases linearly with increasing load.

The high temperature stability study revealed differences in behaviour of the (Pu,Zr)N at temperatures of 2200–2300°C in various environments:

(a) In vacuum, fuel stability was the lowest. Presumably, plutonium metal was evaporated through matrix open porosity.
(b) In an argon and nitrogen environment, the samples revealed high stability. However, separation of phases with high plutonium concentration was observed.

Thermal expansion of ZrN and (Pu,Zr)N samples was investigated under similar conditions by using the dilatometric system at temperatures of 20–1400ºC

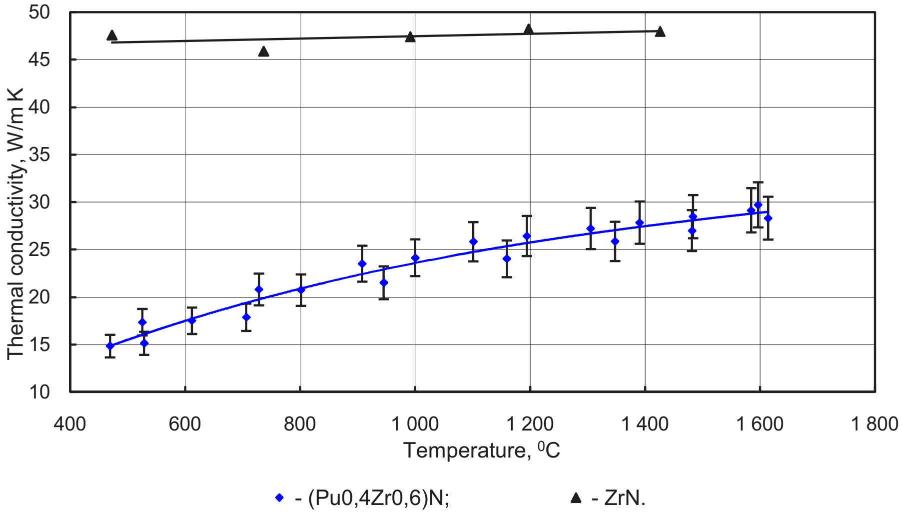

FIG. 3. Temperature dependencies of $(Pu_{0.4}Zr_{0.6})N$ and ZrN thermal conductivity normalized to theoretical density [7].

in the argon plus hydrogen environment. For ZrN, good agreement with literature data has been observed.

The results of thermal conductivity measurements are shown in Fig. 3.

3.5. Minor actinide fuel

In the framework of the collaborative Russian–French AMBOINE project (americium in BOR-60: incineration experiment), the possibility for Am recycling using pyrochemical methods of inert matrix fuel fabrication has been studied [4]. The programme included the fabrication of a BOR-60 experimental fuel pin with vipac $(U,Am)O_2$ in the core and with vipac $(U,Am)O_2$+MgO in the axial blankets. Investigations on Am/rare earth elements and MgO separation by selective precipitation in molten salts have also been carried out.

Through the DOVITA programme, the irradiation of $(U,Np)O_2$ fuel to 20 at.% has been done at the BOR-60 reactor. No principal difference is seen comparing this fuel with MOX or uranium oxide, as shown in Fig. 4 [3].

In the framework of the ISTC MATINE project, the possibility has been considered of fabricating (Pu,Am,Cm,Zr)N fuel containing up to 10 mol% Cm at the RIAR site [4]. Electrolytic refining in the molten chlorides on the liquid metal cathode was proposed as the main flow sheet. Studies have shown the technical feasibility of these processes with respect to RIAR facilities, regulations and technology requirements.

FIG. 4. (U,Np)O_2 at 20 at.% after irradiation at BOR-60.

3.6. Code development

The base code for the performance calculation of cylindrical fuel pins is the KONDOR code developed by IPPE, Obninsk. On the basis of this code, the KORAT code has been developed by VNIINM, Moscow, for the MOX pellet fuel pin calculations and the VIKOND code by RIAR, Dimitrovgrad, for vibro MOX fuel pin calculations. All of these codes realize the calculation of the thermomechanical characteristics of one separately taken axial section of a fuel pin.

With the purpose of minimizing computer resource consumption, the DRAKON-3D code has been developed by IPPE [8]. The modules of the KONDOR code are key ones in the structure of the DRAKON code. The DRAKON code may be used for the temperature and stress–strain state calculations of cylindrical fuel pins with MOX and different types of dense fuel, both in steady conditions and in transients. It considers N_z ~100–200 axial sections of a fuel pin. In the DRAKON code, each functional module represents the independent structure; communication with the surroundings is carried out through the interface parameters. Such an approach permits a consistent increase in the complexity of both the separate modules and the complex as a whole. The general code verification and its modernization become simpler.

The two following models have been developed for estimating the swelling of dense fuel types, e.g. (Pu,An,Zr)N (where An = Am, Cm, Np):

(1) *Model of spherical cells.* The fuel is considered to have a dispersive composition. It is supposed that the dispersed fuel consists of identical cells, regularly located in the fuel volume (according to the face centered cubic

scheme). Each cell represents a thick-walled spherical cladding made of non-fissile (ZrN) material with a spherical grain of fissile material inside.

(2) *Model of spherical gas pores.* The fuel is considered a homogeneous mechanical mixture of ZrN matrix and fissile material. According to this model, the volume of fuel is conditionally divided into regular spherical cells, each of which contains one pore.

The model of spherical gas pores is used for the modelling of (U,Pu)N swelling. Verification of the model has been carried out. The calculation results were compared with the experimental data for nitride fuel pins irradiated in various reactors. The required coefficients have been received [8].

For the performance evaluation of fuel pins under the conditions of a so-called ‘rigid’ loading scheme typical for dense fuels, the correct data on fuel swelling and creep are of principal importance, as well as cladding deformation capability. The BORA–BORA results of the irradiation and post-irradiation examination of four nitride pins are important for the verification and modification of some code modules (temperature, fuel swelling and gas release). An example of hoop stress calculation for the inner surface of $UPu_{0.6}N$ cladding is given in Fig. 5. The first fuel–cladding contact takes place in the middle axial sections with the maximum fuel swelling rate. The stresses in ‘hotter’ sections relax with time, and the maximum stress is realized in the bottom sections with relatively ‘cold’ fuel and an absence of cladding high temperature creep. Currently, the DRAKON code verification is under way. Although the study is not yet complete, the first results show that the code permits the calculation of temperature, stresses and deformation of mixed nitride pins with some margin.

4. CONCLUSION

Different fuels have been studied in BN reactors (i.e. PuO_2, UO_2, (pellet, vibro), $(U,Pu)O_2$ (pellet, vibro), UC, UN, (U,Pu)C, (U,Pu)N, oxide, nitride and carbide inert matrix fuels, as well as alloyed and non-alloyed metallic fuels). Recently, experiments with (U,Pu)N, MgO and ZrN based fuels have been completed in the BOR-60.

In order for fast sodium reactors to meet the requirements related to nuclear installations of the fourth generation, studies on sustainability, proliferation resistance, waste management, safety, economics and R&D of the commercial BN-K reactor are currently under way in the Russian Federation. As a reference fuel, MOX fuel is considered to be a better long term option than nitride is assumed to be. Other types of dense fuel are planned to be studied as well.

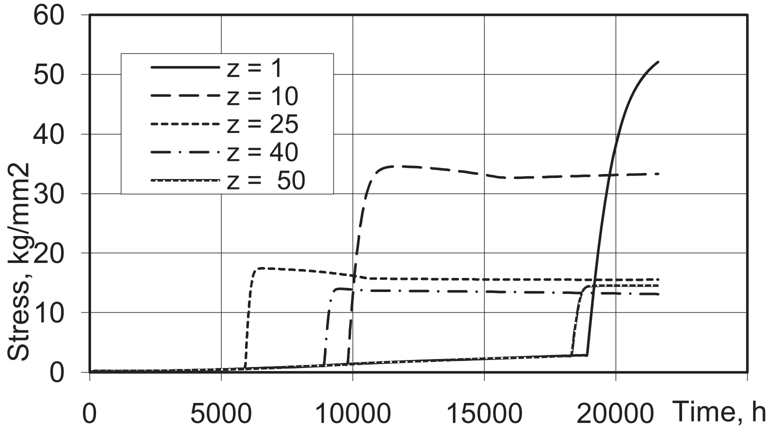

FIG. 5. Hoop stress versus time in different axial sections (z) of a $UPu_{0.6}N$ pin.

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ADVANCED FUELS FOR FAST REACTORS*

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

FAST REACTOR FUEL DEVELOPMENT IN EUROPE

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Abstract

Research and development of minor-actinide-bearing fuels in Europe has made significant progress, with a number of scoping irradiation tests made on a number of candidate fuels foreseen for fast reactors and dedicated minor actinide transmutation systems, e.g. the accelerator driven system. Currently, efforts concentrate on uranium based fuels, as the deployment of fast reactor fleets requires Pu generation in order to achieve sustainability. Both homogeneous and heterogeneous concepts for minor actinide reactor recycling are considered. In the former, the minor actinides are added in small quantities to the mixed oxide fuel, while in the latter, the minor actinides are loaded in significant quantities in UO_2. Irradiation programmes to test these concepts for pellet and SPHEREPAC fuel configurations are under way.

1. INTRODUCTION

The advent of the Generation IV International Forum (GIF) has given a renewed impetus to the development of fuels for fast reactor applications. Europe has a rich tradition in this area and five fast reactors (Rapsodie, Phénix, Superphénix, KNKII and DFR) have been operated successfully. Mixed oxide (MOX) consisting of $(U,Pu)O_2$ was the fuel of choice for these systems, all of which were sodium cooled. Despite the convergence on oxide fuel, large R&D programmes were dedicated to carbide and nitride fuels, as they enable higher fissile density to be achieved, improving core neutronic and safety performance. They also have a higher thermal conductivity than oxide fuels, providing a greater margin between the operating and melting temperatures.

The Generation IV initiative clearly requires waste minimization that can be achieved by recycling and transmutation of minor actinides (MAs). It is this aspect of fast reactor fuel R&D that is presented here. A chronological summary of past programmes with MA fuels is discussed. Fuels for sodium, lead and gas cooled fast reactors (SFRs, LFRs and GFRs, respectively), as well as the dedicated MA transmuter, the accelerator driven system, have been fabricated and irradiated.

2. SUPERFACT

Following the irradiation experiment FACT, an important milestone irradiation of MA fuels in the Phénix reactor was undertaken in a joint programme by the Commissariat à l'Énergie Atomique (CEA) and the Joint Research Centre–Institute of Transuranium Elements (JRC-ITU). This was the first fast reactor test of fuels and targets characteristic of the homogeneous and the heterogeneous MA recycle processes. The fuels and targets were:

$(U_{0.74}Pu_{0.24}Am_{0.02})O_{1.973}$
$(U_{0.74}Pu_{0.24}Np_{0.02})O_{1.973}$
$(U_{0.60}Np_{0.21}Am_{0.19})O_{1.926}$
$(U_{0.55}Np_{0.45})O_{1.996}$

All fuels and targets contained U, as was foreseen in a strategy whereby breeding gains greater than unity are sought. The fuels were manufactured at JRC-ITU using a sol gel–route, which has the advantage (in contrast to powder metallurgy) of giving a perfect solid solution, an ideal starting point for such scoping studies. Pellets (see Fig. 1) with about 95% of the theoretical density, were loaded in standard Phénix stainless steel grade AIM1 cladding and were sealed with 1 bar He. A total of 8 pins (two for each fuel) were loaded in a MOX assembly of the Phénix reactor to ensure the most representative neutron spectrum and to enable a direct performance comparison with qualified Phénix MOX fuel to be made. The fuel pins were irradiated for 382.5 equivalent full power days (EFPD).

Excellent irradiation behaviour was found for the pins. The linear power of the Pu-bearing fuel decreased from 380 to 320 W/cm during irradiation, while the dedicated target concept fuels showed an increase in power from 170 to 280 W/cm during irradiation [1]. Post-irradiation examination (PIE) was performed at CEA and JRC-ITU hot cells. The characteristic central hole was

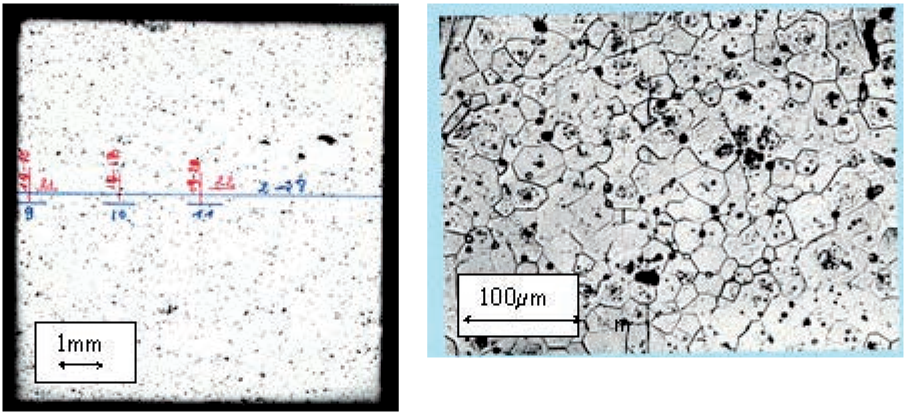

FIG. 1. Ceramographs of the $(U_{0.74}Pu_{0.24}Am_{0.02})O_2$ *fuel irradiated in the SUPERFACT experiment in Phénix.*

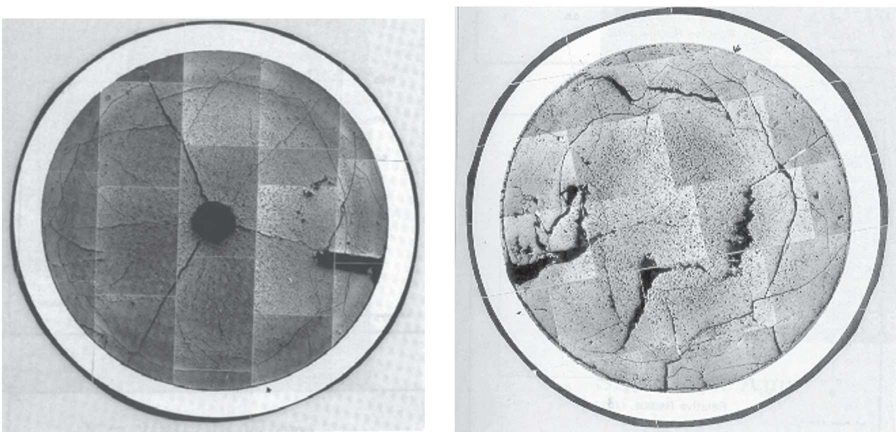

FIG. 2. Ceramographs of $(U_{0.74}Pu_{0.24}Am_{0.02})O_{1.973}$ *fuel and* $(U_{0.80}Np_{0.21}Am_{0.19})O_{1.926}$ *target irradiated in the Phénix reactor (SUPERFACT experiment).*

formed in the Pu-bearing fuels (see Fig. 2). Fission gas release rates (60–80% of that produced) were in good agreement with those of standard MOX fuels deployed in the same assembly, even for the fuels with a high concentration of MAs. No major difference between these and the standard MOX fuels was observed. Higher helium generation and release in the $(U,Pu,Am)O_2$ fuel was noted. A significant swelling of the $(U,Np,Am)O_2$ fuel pin was detected and was attributed to helium buildup in the fuel. Furthermore, this pin showed a high helium production which may have contributed to the slight fuel swelling. The efficiency of transmutation in all fuels was about 30% [2].

3. EFTTRA-T4

During the 1990s, the emphasis switched from Pu breeding to Pu and MA destruction, which is a major goal of the accelerator driven system. Thus, fuels and targets bearing fertile ^{238}U were not favoured and the inert matrix fuel concept was deployed. Various materials were selected on the basis of their neutronic inertness, their actinide, coolant and cladding compatibility, and their thermophysical and thermochemical properties.

Magnesium aluminate spinel ($MgAl_2O_4$) loaded with 0.4 g/cm^3 Am was the first candidate to be tested, c.f. the EFTTRA-T4 irradiation test [3, 4]. The fuel was manufactured by JRC-ITU, by infiltration of a porous $MgAl_2O_4$ pellet with Am nitrate solution, which was then sintered to reach 95% theoretical density. The fuel was a ceramic–ceramic (CERCER) composite, consisting of $AmAlO_3$ particles [5], 1–2 μm in diameter, dispersed throughout the matrix. The Am distribution was not even and a shell with higher (14 wt% versus 9 wt%) enrichment was present in the pellet. Two pins were irradiated in the HFR Petten for 358 and 652 EFPD, respectively. A dramatic volumetric fuel swelling of 18 and 27%, respectively, was observed. This swelling was mainly caused by helium, which, due to the low irradiation power and temperature, was retained in bubbles in the fuel (see Fig. 3). This result clearly demonstrated that its generation and retention in MA-bearing fuels could have a detrimental effect vis á vis fuel performance and due account must be made in the design of the fuel (operating temperature, porosity, etc.).

4. ECRIX

The CEA focused on MgO as a primary candidate to form an inert matrix for MA transmutation. Unlike spinel, no chemical interaction occurs with actinide oxides. Furthermore, the thermal conductivity of MgO is higher than UO_2 and MgO is soluble in nitric acid under PUREX conditions, so that such MgO–AmO_2 composites are readily reprocessable. The MgO–$AmO_{1.62}$ fuels with an Am loading of 0.7 g/cm^3 were fabricated using a powder metallurgy process which yielded a composite with fine (<30 μm) particles of $AmO_{1.62}$ dispersed in the MgO matrix. Two fuel pins were irradiated in specially adapted capsules (ECRIX-H and ECRIX-B) in the Phénix reactor. Both capsules used a moderator (CaH_2 and $^{11}B_4C$, respectively) to moderate the neutron spectrum, which should enhance the transmutation efficiency.

PIE of the ECRIX-H fuel pin has been performed in the LECA Cadarache and Atalante Marcoule facilities. In contrast to EFTTRA-T4, no major swelling was encountered. The average decrease in the geometrical density of the pellets after irradiation was close to 6.7% [6]. Helium release was not high (around 23%

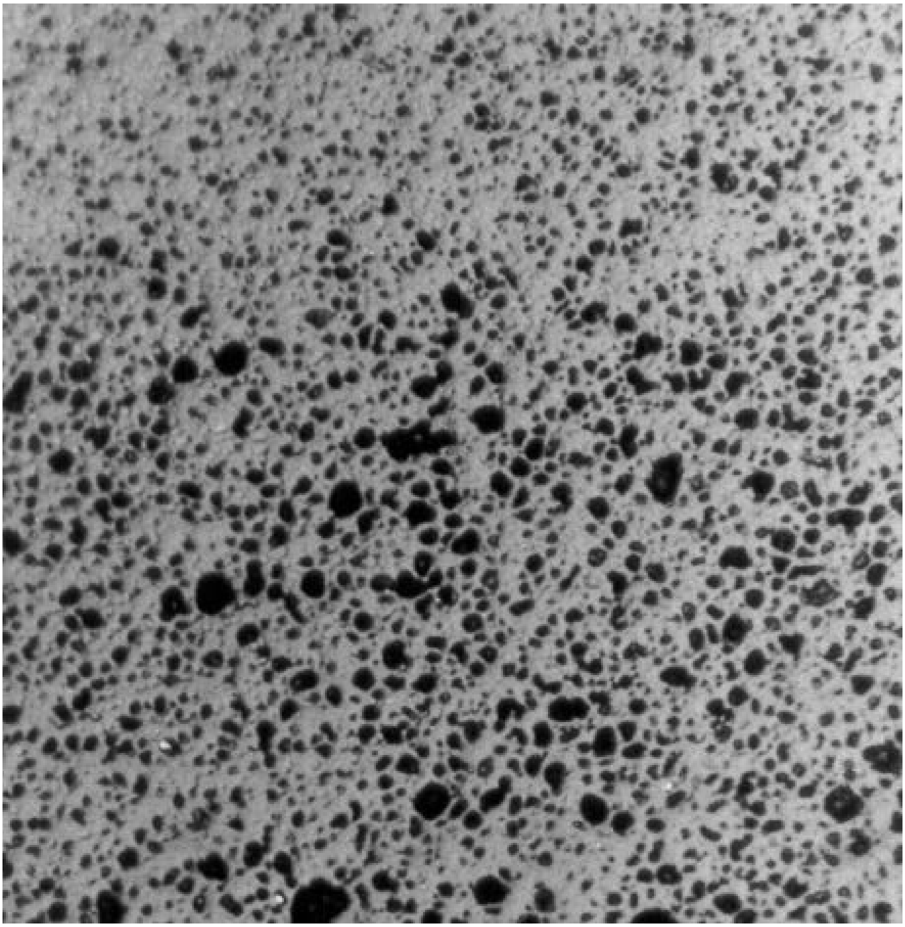

FIG. 3. Helium bubbles observed in the irradiated EFTTRA-T4 fuel.

of the He produced), which suggests that a large part of the He is stored in the MgO matrix. Thermal treatment carried out at high temperature (up to 2000°C) on irradiated samples confirmed the high content of He trapped in the composite pellets after irradiation and cooling. The ^{241}Am fission and transmutation rates were ~30 at.% and 95 at.%, respectively, after an irradiation period of 318 EFPD. Ceramographic examination (see Fig. 4) revealed some cracks in pellets (few large fragments per pellet) and a relatively high porosity.

The previous $AmO_{1.62}$ particles of the composite pellets contained mainly Pu after irradiation. Fission products are mostly implanted in the MgO matrix. Globally, rather good behaviour of the ECRIX-H targets was observed under irradiation up to an Am transmutation rate close to 95 at.%. PIE of these pins is still ongoing.

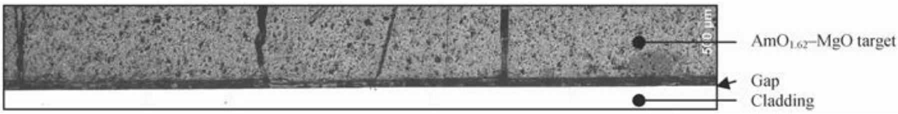

FIG. 4. Ceramographic examination of a section of the ECRIX-H pin.

5. CAMIX/COCHIX

The negative results of the EFTTRA-T4 test led to the quest for a material which could readily incorporate MAs into its crystal structure. Yttria stabilized zirconia (YSZ) has the same crystallographic structure as UO_2 and is therefore an ideal candidate. It has also been used in parallel investigations for the burning of Pu in light water reactors. Its disadvantages lie in its insolubility in nitric acid and its low thermal conductivity. The latter can be countered by using composite CERCER or CERMET concepts. Besides, a low thermal conductivity can favour a high release of fission gas and helium and consequently lead to a low swelling of the pellets, as the irradiation temperature will be higher.

The CAMIX and COCHIX experiments comprise $(ZrY,Am)O_2$ (CAMIX 1) and MgO–$(Zr,Y,Am)O_2$ (CAMIX 2 and COCHIX 3) targets with Am loadings of 0.7 g/cm^3. In the CAMIX 2 and COCHIX 3 pellets, the Am in the YSZ phase was increased to enable a 13 vol.% dilution with MgO. Furthermore, the $(Zr,Y,Am)O_2$ phase was size selected to give micro- and macrodispersions (see Fig. 5). In the former, complete damage of the MgO matrix is expected, while in the macrodispersion, damage should be restricted to non-overlapping ~10 μm shells around the fissile particles. Thus, the properties of the MgO matrix itself are only affected by neutron irradiation, posing less severe property degradation, as was shown in the MATINA 1A programme in Phénix [7]. The main characteristics of the three types of target – fabricated at ITU and CEA – are given in Table 1.

The estimated maximum operating temperatures are around 2200°C for CAMIX 1 and 1400–1500°C for CAMIX 2 and COCHIX 3 pellets. CAMIX 1 was irradiated for 234 EFPD, while CAMIX 2 and COCHIX 3 pins were withdrawn from the reactor after 178 EFPD. The corresponding neutron fluences were ~1.9×10^{26}/m^2 and 1.45×10^{26}/m^2, respectively, (with $E > 0.1$ MeV), while the expected ^{241}Am fission rates are ~23 and 17 at.%, respectively. The maximum linear power was 80–100 W/cm. The irradiation of these pins was completed in March 2009 and PIE is awaited.

FIG. 5. Macrograph of a COCHIX 3 pellet.

TABLE 1. CAMIX–COCHIX PELLET COMPOSITIONS

Pin	Material	Fuel form
CAMIX 1	$Am_{0.06}Zr_{0.78}Y_{0.16})O_{2-x}$	Solid solution
COCHIX 2	$MgO–(Am_{0.2}Zr_{0.66}Y_{0.14})O_{2-x}$	Microdispersed 50 µm
COCHIX 3	$MgO–(Am_{0.2}Zr_{0.66}Y_{0.14})O_{2-x}$	Macrodispersed 100 µm

6. FUTURIX FTA

This is a multinational programme involving the United States Department of Energy, the Japan Atomic Energy Agency, ITU and CEA which is dedicated to MA-bearing fissile (Pu based fuel) and low fertility fuels [8]. In the frame of this programme, the CEA and JRC-ITU have provided CERCER and CERMET fuels, respectively, and the United States Department of Energy provided metallic and nitride fuels. The CEA used an oxalate precipitation procedure to produce $(Pu,Am)O_2$ with 50% and 80% Am fissile phases, which were then dispersed in MgO at 20 and 25 vol.%, respectively. The fissile particle size range lay between ~1 and 100 μm. The JRC-ITU fuels were based on Mo as a matrix. The ceramic phase was $(Pu,Am)O_2$ with 20% Am and $(Zr,Pu,Am)O_2$ with roughly equal quantities of Pu and Am. Compared to previous experiments (ECRIX, CAMIX, COCHIX, EFTTRA-T4), the Am content in the FUTURIX FTA fuel is particularly high (see Table 2) and should lead to a very high production of helium during irradiation.

The fuel pins fabricated by the CEA and ITU (see Figs 6 and 7) were irradiated under a fast neutron flux in the Phénix reactor for 234 EFPD and PIE should start in 2010. Important information is expected from this irradiation experiment and in particular regarding the behaviour of the MgO and Mo matrices with respect to very high production of He.

7. HELIOS

Helios is a dedicated separate effect study on the influence of helium in transmutation fuels. Five fuel pins are currently under irradiation in the HFR Petten. CEA produced a MgO based CERCER with $Zr_2Am_2O_7$ as the fissile phase. The Am loading was 0.7 g/cm^3 for this and the four remaining fuels. The pyrochlore superstructure of $Zr_2Am_2O_7$ is cubic with a lattice parameter roughly twice that of the cubic stabilized zirconia. Two other fuels were based on YSZ and were prepared with (0.39 g/cm^3 Pu) and without Pu, giving similar fuels but operating at different linear powers and, of course, temperatures, which will culminate in different He release behaviour. The final pair of fuels were Mo based CERMET with $(Pu,Am)O_2$ and $(Zr,Pu,Am)O_2$ as the fissile phase. The irradiation test will end at the beginning of 2010. Examples of pellets are shown in Fig. 8.

TABLE 2. COMPOSITION OF THE FUELS PROVIDED BY CEA AND ITU

Fabricator	Target composition	Am content (g/cm^3)	TRU (g/cm^3)
CEA	$(Pu_{0.2}Am_{0.8})O_2$ + 75 vol.% MgO	1.9	2.4
	$(Pu_{0.5}Am_{0.5})O_2$ + 80 vol.% MgO	1.0	2.0
ITU	$(Pu_{0.23}Am_{0.24}Zr_{0.53})O_2$ + 60 vol.% Mo	1.0	1.8
	$(Pu_{0.8}Am_{0.2})O_2$ + 86 vol.% Mo	0.3	1.3

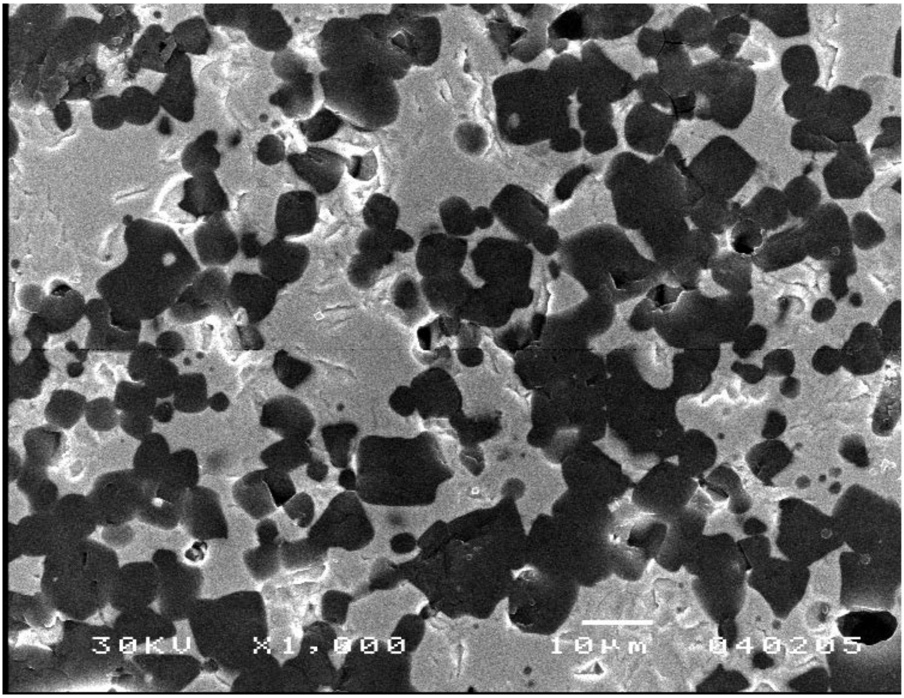

FIG. 6. Micrograph of the $(Pu_{0.2}Am_{0.8})O_{2-x}$ MgO CERCER pellet.

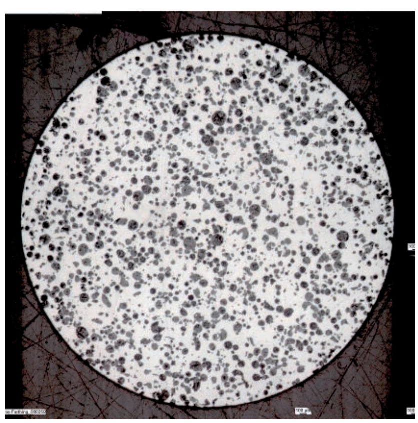

FIG. 7. Ceramograph of $(Pu_{0.23}Am_{0.24}Zr_{0.53})O_{2-x}$ + 60 vol.% Mo CERMET pellet.

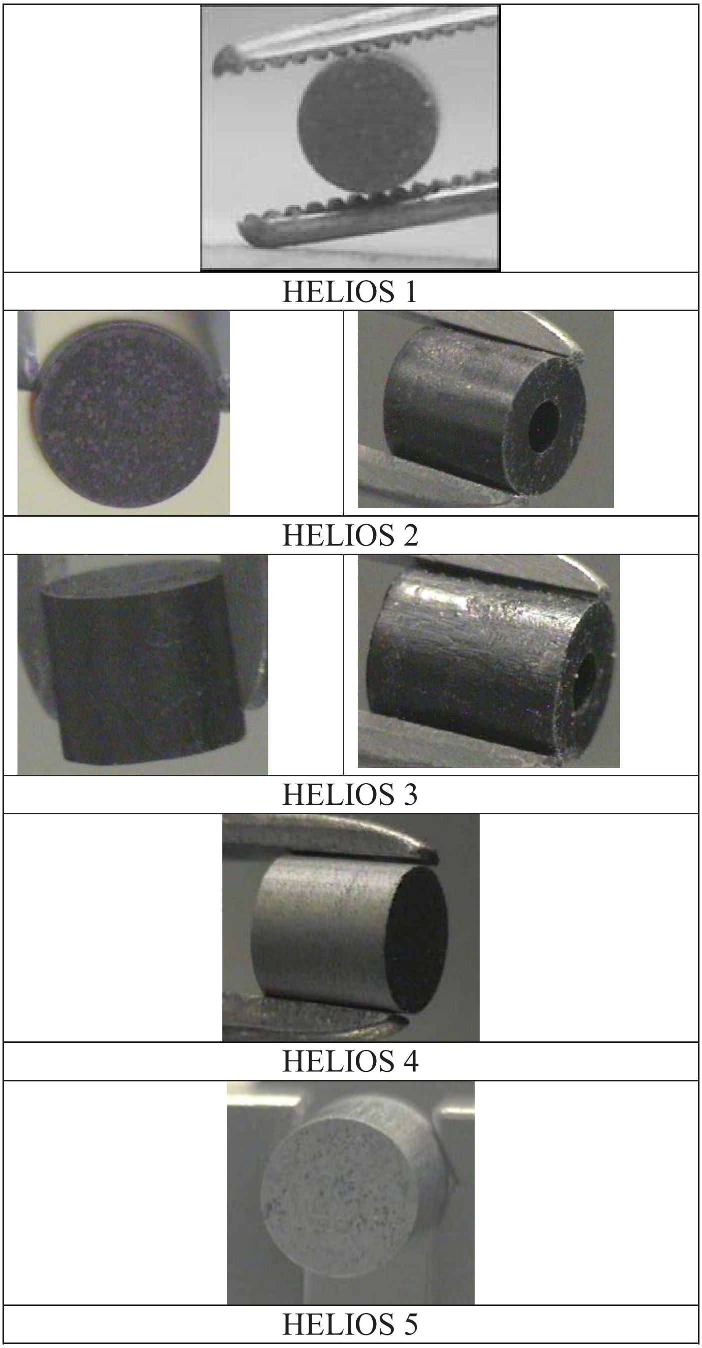

FIG. 8. Pellets from the HELIOS experiment.

8. IRRADIATION TESTS IN PREPARATION

Currently, attention has returned to fertile based fuels, largely in support of the fast reactor programme in France, where the ASTRID reactor is planned for operation in 2020. The CP ESFR project foresees fabrication of carbide and nitride fuels with low Am contents and determination of their vaporization behaviour. In an additional programme (FAIRFUELS), two irradiation tests are in preparation. The MARIOS programme will investigate He release in $(U,Am)O_2$. The fuel will be in the form of disks wedged between densimet (a tungsten alloy) heat sinks, so that temperature gradients are minimized in the fuel. SPHERE addresses He management issues and compares the performance of $(U,Pu,Am)O_2$ fuel in pellet and SPHEREPAC configurations.

9. OUTLOOK

MA-bearing fuels have been studied in Europe, but only to a limited extent, i.e. no more than scoping studies have been performed. Progress has been made and the properties of fresh fuels have been determined without, or in association with, dedicated irradiation experiments. Progress has been influenced by political environments and today the milestone experiment SUPERFACT, along with the recent AM1 test in Japan, are foremost in our understanding of fertile based MA fuels and targets. Clearly, this is not enough and detailed fuel studies are required for both normal and off-normal operation. Indeed, nothing at all is known about the behaviour of MA-bearing fuels in off-normal reactor operating conditions.

Development of transmutation fuels can only be achieved through international collaboration (e.g. Generation IV). Furthermore, within Europe, a Sustainable Nuclear Energy Technology Platform (SNETP) has been established to facilitate and guide European research efforts for nuclear energy implementation in the next century. SNETP has recently published a strategic research agenda and clearly defines R&D needs for fast reactor fuels. For fuel R&D, the following needs have been recognized:

- Fabrication and basic property determination (heat capacity, thermal conductivity, vaporization behaviour, coolant–cladding compatibility, etc.) must be made on fresh and irradiated materials.
- Dedicated irradiation experiments, designed to establish the influence of specific effects (He buildup and release, etc.), as well as basic performance data, are needed to establish feasibility, recognize potential ‘showstoppers’ and elucidate specific effects.

- Resource limitations must be recognized and compensated as much as possible through theoretical studies, along with simulation and modelling, to provide not only the best possible understanding of the chemistry and physics underpinning the processes ensuing as a consequence of fission, but also to predict and design the most relevant experiments to establish key irradiation performance data. Normal conditions and accidental situations have to be considered in this approach.
- Facilities must be built that are capable of fabricating industrial scale quantities of MA fuel and representative fuel pins and bundles.
- Qualification in representative fast neutron spectra is an absolute necessity, as is dedicated off-normal testing.

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RECYCLE STRATEGIES FOR FAST REACTORS AND RELATED FUEL CYCLE TECHNOLOGIES

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Abstract

Fast reactors and related fuel cycle (hereafter referred to as 'fast reactor cycle') technologies have the potential to contribute to long term energy security owing to their effective use of uranium and plutonium resources, and to a reduction in the heat generation and potential toxicity of high level radioactive wastes by burning long lived minor actinides recovered from spent fuel from light water reactors and fast reactors. Further, it is likely that fast reactor cycle technologies can play a certain role in non-proliferation as addressed in the Global Nuclear Energy Partnership. With these features, the research and development towards their commercialization has been promoted vigorously and globally as a future vision of nuclear energy. The introduction of fast reactor cycle systems will be carried out independently in each country according to its national conditions and nuclear energy policy. It should then be considered important to have a globally common consensus relating to safety philosophy, concepts of proliferation resistance, transuranic element burnup and recycling and so on. For the development and utilization of fast reactor cycle systems, while respecting each country's concept, it is essential to organize the technologies and concepts which countires should have in common globally and build a framework to make them standardized. The use of existing frameworks such as the Generation IV International Forum and the International Project on Innovative Nuclear Reactors and Fuel Cycles is considered effective to achieving this. Furthermore, a vigorous promotion such as international cooperative developments enables the formation of international consensus on major technologies for the fast reactor cycle as well as the saving of resources by infrastructure sharing.

1. INTRODUCTION

World energy consumption tends to increase because of the increase in the world's population and in energy consumption per capita due to economic growth, with a central focus on developing countries. Energy security is recognized as an extremely important issue in all the countries of the world. Further, it should also be considered that the emission of greenhouse gases increases and climate change caused by global warming is an issue the world is facing today.

Many countries are considering introducing nuclear energy for the first time and the expansion of nuclear energy is planned in many countries because only nuclear power does not emit greenhouse gases among the large scale power generation technologies. Therefore, some problems such as the restriction of uranium resources, the management of radioactive waste and nuclear proliferation are topical [1–4].

Because fast reactors (FRs) and related fuel cycle (hereafter referred to as the 'FR cycle') technology can solve such energy security issues fundamentally by drastically enhancing the efficiency of utilization of uranium resources, the development of FR cycle technology has been advanced in some countries since the dawn of nuclear development. In addition, because this FR cycle technology can burn long lived minor actinides recovered through reprocessing spent nuclear fuels from LWRs and FRs, it has recently been re-evaluated from the viewpoint of environmental impact as a technology that can decrease heat generation and a potential environmental burden (radioactivity toxicity) posed by high level radioactive waste [1]. Therefore, many countries are aggressively advancing research and development (R&D) for the commercialization of FR cycle technology, indispensable to the future expansion of nuclear energy. In this case, sufficient consideration of safety is necessary as a common issue. Moreover, because FR cycle technology involves a large amount of plutonium (Pu) in its system, special consideration of nuclear non-proliferation is necessary.

It is expected that FR cycle technology can solve both energy resource and environmental issues harmoniously for the sustainable future development of society. However, the fact is that considerable resources are required for the development of FR cycle technology. Further, multilateral cooperation is essential to carry out the development because some issues such as safety and nuclear non-proliferation cannot be solved without an international consensus. It is, therefore, thought that international cooperation will become more important in the future.

The fuel cycle is indispensable for a long term stable supply of energy; the development of FR cycle technology that is consistent with the fuel cycle enables FRs to fulfil their potential. In this paper, the development situation of FR cycle technology in each country and international cooperation are reviewed. Common issues to advance FR cycle technology development steadily are highlighted, based on worldwide development trends, and a coordinated strengthening of international cooperation is proposed as the solution.

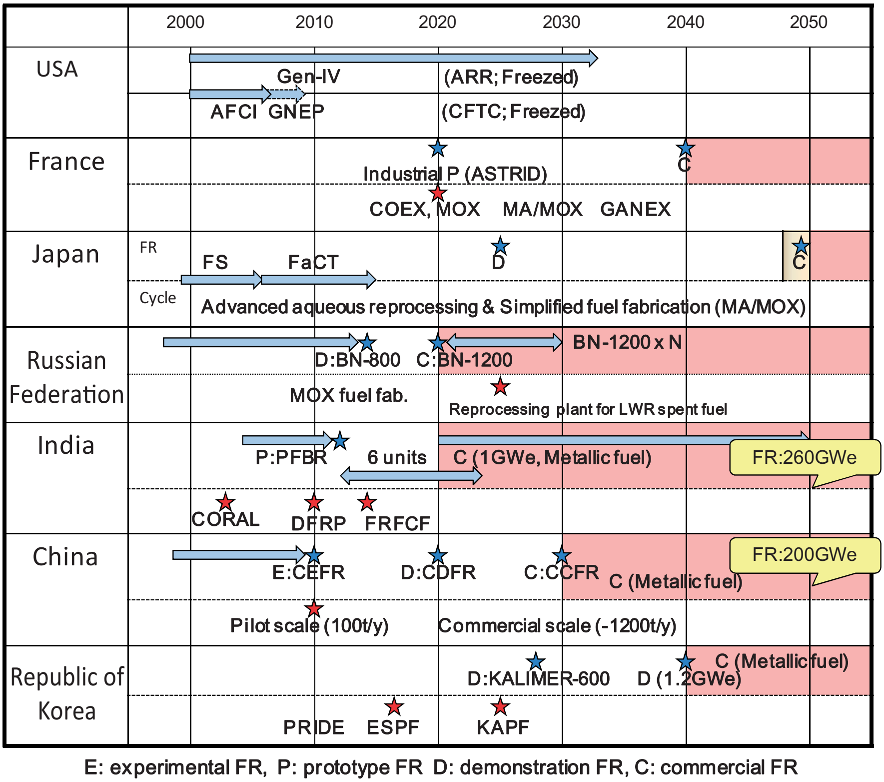

FIG. 1. FR cycle national development plan.

2. RECYCLE STRATEGIES WITH FRs IN EACH COUNTRY

China, France, India, the Republic of Korea, the Russian Federation and the United States of America also aim to develop the closed fuel cycle using FRs with oxide fuel or metallic fuel as described below. Figure 1 shows the development plan of the FR cycle envisioned by each country.

2.1. Japan

In Japan, Pu recovered through reprocessing LWR spent fuels is currently used in LWRs, and this will continue until FR cycle technology has been established in the future. It is important how it shifts from utilization of Pu in LWRs to use in FRs [5]. The five party coordinate council, including Government authorities, comprising MEXT (Ministry of Education, Culture, Sports, Science and Technology), METI (Ministry of Economy, Trade and Industry), utilities, vendors and the JAEA (Japan Atomic Energy Agency) was established in July 2006, and it is now discussing how to shift smoothly from the existing LWR cycle to the FR cycle [6].

In the Feasibility Study on Commercialized Fast Reactor Cycle Systems (FS) executed from July 1999 to the Japanese fiscal year 2005, the candidate concepts for various categories of FR, reprocessing and fuel fabrication were examined and evaluated for the adaptability and technical viability of development goals such as safety, economic competitiveness, reduction of environmental burden, effective use of resources and nuclear proliferation resistance. As a result, a combination of sodium cooled FR (SFR), advanced reprocessing and simplified pelletizing fuel fabrication of minor-actinide-bearing mixed oxide (MOX) fuel was selected as a main concept due to the reason that it is the most promising and enables early practical use. Then, as an alternative, a combination of SFR, electrorefining and injection casting of minor-actinide-bearing metallic fuel was selected because metallic fuel can improve core performance further, compared with MOX fuel, in the case where the future demand and supply of uranium might be more problematic than expected [7].

In the Fast Reactor Cycle Technology Development (FaCT) project, which began in 2006, the design studies and the R&D of innovative technologies (including 13 items related to the reactor, 6 items related to the reprocessing and 6 items related to the fuel fabrication) have been advanced, based on the above-mentioned main concept. It is assumed that approval of the innovative technologies as a whole system can be ascertained and that the conceptual design of both demonstration and commercial facilities, as well as the R&D plans for practical use, can be presented in 2015 [8–10].

Japan has an experimental FR (Joyo) and a prototype FR (Monju). It is assumed that data necessary for future development can be accumulated from these reactors, combined with achievements of the FaCT project. It is then expected that a demonstration FR and its fuel cycle facility can start operating around 2025 and, with further operating experience, a commercial FR can be introduced before 2050 [11].

2.2. France

In France, a demonstration FR, Super-phenix, was shut down in 1998 and a prototype FR, Phenix, is scheduled to be shut down this year. France is now developing an industrial prototype Generation IV FR, selecting one concept from either the SFR or gas cooled FR (GFR), aiming at becoming operational in 2020. The tentative specification was planned to be fixed in 2009 and the technological specification in 2012. It is reported that the SFR, named ASTRID, will be adopted as a prototype FR in 2020, and the GFR would be a long term alternative. The Generation IV FR is scheduled to be commercially introduced around 2040 [12].

In France, based on the waste management act of 28 June 2006 (LOI no. 2006-739 du 28 juin 2006 de programme relative à la gestion durable des matières et déchets radioactifs), R&D on partitioning and transmutation technology has been executed, aimed at evaluating and making a decision on fuel cycle technology in 2012. Candidates for recycling technology include COEX, which co-extracts U and Pu, DIAMEX-SANEX, which separates minor actinides, and GANEX, which extracts transuranic elements in the 'lump'. The MOX fuel fabrication facility known as AFC for the ASTRID core is planned to be constructed before 2020. Further, the minor-actinide-bearing fuel fabrication facility known as ALFA is under consideration for demonstrating the minor actinide transmutation capability of ASTRID [13].

2.3. USA

In the USA, assuming that Pu extracted by reprocessing may pose problems with regard to the proliferation of nuclear weapons, a plan for commercial reprocessing has been suspended and FR development has likewise been suspended, although an FR was operational after 1977. The former Bush administration supported deployment of nuclear energy from the viewpoint of emphasizing the improvement of energy security. They referred to the development promotion of nuclear fuel cycle technology and next generation nuclear power technology in the National Energy Policy, announced in 2001. The Advanced Fuel Cycle Initiative (AFCI) was started in response to the policy. In a report of the Government concerning the AFCI in 2003, it was proposed that the following be concurrently executed [14]:

- Intermediate term issues associated with spent nuclear fuel, specifically reducing the volume of material requiring geological disposition by extracting uranium (which represents 96% of the constituents of spent nuclear fuel), and reducing the proliferation risk through the destruction of significant quantities of Pu contained in spent nuclear fuel;
- Long term issues associated with spent nuclear fuel, specifically the development of fuel cycle technologies that could sharply reduce the long term radiotoxicity and long term heat load of high level waste sent to a geological repository.

The USA announced the Global Nuclear Energy Partnership (GNEP) in 2006, in which R&D has been promoted, aimed at the commencement of commercial operation of the advanced recycling reactor and the Consolidated Fuel Treatment Center which will enable actinide recycling in the 2020s. However, the development of these plants was suspended due to regime change in January 2009 and to nuclear policy shifts in direction which emphasized long term scientific R&D. However, the AFCI is still ongoing as an advanced fuel cycle and waste management technology with proliferation resistance, aimed at the minimization of nuclear waste. In addition, the R&D of Generation IV reactors which will enhance safety, cost effectiveness and proliferation resistance of nuclear power is advanced [15].

2.4. Russian Federation

On the basis of considerable operational experience gained over 140 reactor-years with experimental SFRs (BOR-60, etc.) and prototype reactors (BN-350 and BN-600, UO_2 fuel), a BN-800 (MOX fuel) has been built in the Russian Federation, with commissioning aimed at 2014. Later, a pilot plant BN-1200 is scheduled to start operating in 2020 and a small series of BN-1200s, as commercial reactors, are also planned to be deployed by 2030. In addition, with abundant experience of lead and lead–bismuth cooled reactors, which are used as the power source of a nuclear submarine, the Russian Federation is executing conceptual designs for the lead cooled FR (LFR) BREST and the lead–bismuth cooled FR SVBR-100, which can enhance safety and proliferation resistance [16–18].

A closed fuel cycle with FRs (BN-1200) is pursued in which the minor actinide transmutation issues are under discussion. A MOX fuel fabrication facility for FRs is currently designed to start using Pu recovered from Russian PWR (WWER) spent fuels. A large scale reprocessing plant for LWR spent fuels is planned to start operation around 2025. Further, technologies for electrowinning and vibropacked fuel fabrication are being developed, and irradiation tests for fabricated fuels are being carried out using BOR-60 and BN-600 units [16, 19].

2.5. India

India, which has abundant thorium resources, institutes its own three stage strategy aimed at efficient utilization of thorium (the first stage: produce Pu by heavy water reactors; the second stage: FR cycle with MOX fuels produce ^{233}U from thorium fuel blankets of FRs; the third stage: thorium recycling with ^{233}U/Th fuel and advanced heavy water reactors). Currently, India is advancing the development of the FR cycle in the second stage [20].

An experimental SFR (FBTR, carbide fuel) has been operating since 1985 in India. Now, a prototype SFR (PFBR, MOX fuel) is being constructed and is scheduled to begin commercial operation in 2012. Later, three sets of twin plants will start operating with isomorphic reactors by 2023, which have been improved in terms of safety and economics. After 2020, commercial SFRs of 1 GW(e) with metallic fuel will be introduced one by one (because its doubling time for metallic fuel is shorter than that for MOX fuel) to respond to the rapid increase in domestic electricity demand [20]. India has planned to cover 260 GW(e) out of the 275 GW(e) nuclear power capacity with SFRs installed by 2052 [21]. Currently, the capability predicted to be installed around 2050 is being re-evaluated on the basis of the import of LWRs (40 GW(e)) in the future [22].

India has been developing fuel cycle technologies consistent with reactor development and has operated small scale hot testing facilities for reprocessing since 2003. A demonstration reprocessing plant, whose capacity is 1 t/a, is under construction to prove the reprocessing of PFBR fuel. In addition, the Fast Reactor Fuel Cycle Facility that can reprocess, fabricate fuels and process high level waste for three 500 MW(e) FRs is under construction, adjacent to the PFBR site. R&D on the dry reprocessing of metallic fuel has also been executed concurrently [22].

2.6. China

In China, an experimental SFR, which plans to use UO_2 fuel at first and then shift to MOX fuel, is under construction and aims to be in operation in 2010. Further, China plans to start a demonstration SFR with MOX fuel by 2020. China signed an agreement with the Russian Federation in October 2009 to begin a prior project and design work for the construction of two commercial SFRs of 800 MW(e).The introduction of commercial SFRs is planned to begin around 2030 and it is expected that all the commercial reactors to be introduced after around 2050 will be SFRs. It is also scheduled that 200 GW(e) out of 240 GW(e) installed nuclear power capacity should be covered with SFRs by 2050. China, as with India, also plans to shift from MOX fuel to metallic fuel in the future [23].

China is constructing a reprocessing pilot plant (100 t/a) for PWRs and a MOX fuel fabrication plant (0.5 t/a). A commercial reprocessing plant (~1000 t/a), a MOX fuel fabrication plant (2 × 50 t/a) and a MOX fuel reprocessing plant (50 t/a) are planned in the future. Metallic fuel has also been developed [23, 24].

2.7. Republic of Korea

In the Republic of Korea, an SFR (Korea Advanced Liquid Metal Reactor (KALMER)) using metallic fuel has been developed, which is aimed at the effective use of uranium resources and the decrease in high level waste. A demonstration SFR with metallic fuel is scheduled to start operating in 2028. It is then planned to replace the existing PWRs with SFRs by around 2040 [25].

The Republic of Korea is currently establishing a pyroprocess integrated inactive demonstration facility (PRIDE, 10 t/a), a mock-up facility for pyroprocessing, and has plans to start operating an engineering scale pyrochemical process facility (ESPF, 10 t/a) by 2016 and the Korea advanced pyroprocess facility (KAPF, 100 t/a) by 2025. It is assumed in the Republic of Korea that dry processing excels in nuclear proliferation resistance because it cannot isolate pure Pu from spent metallic fuel [25].

3. MULTINATIONAL COOPERATION ON FUTURE NUCLEAR SYSTEMS

Up to 2000, international cooperation concerning R&D of FR cycle technology was chiefly limited to bilateral cooperation, excluding the IAEA's Technical Working Group on Fast Reactors (TWG-FR) and Technical Working Group on Nuclear Fuel Cycle Options (TWG-NFCO). The Generation IV International Forum (GIF), a framework for multilateral cooperation between countries initiated by the USA, started in 2000. Subsequently, the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was set up by the IAEA in 2001 and GNEP by the USA in 2006. The member countries or organizations participating in the multinational cooperation shown in Fig. 2 are interrelated within these cooperative frameworks according to the purpose of the activities.

3.1. GIF

GIF was formed in 2000 by the United States Department of Energy for examining advanced nuclear energy system concepts. GIF consists of twelve member countries and one international organization which signed the Charter. Six systems, SFR, GFR, LFR, molten salt reactor, surpercritical water reactor and very high temperature gas cooled reactor are selected as its cooperative research subjects and of these, three systems (SFR, GFR, LFR) are FRs. The other two systems (molten salt reactor, supercritical water reactor) can possibly be used as FRs. Three Methodology Working Groups are responsible for developing and

IAEA Member States and international organizations participating in each framework of international cooperation (as of December 2009)

	GNEP	GIF	INPRO	TWG-FR	TWG-NFCO
Asia	China Japan Republic of Korea	China Japan Republic of Korea	China India Indonesia Japan Republic of Korea Pakistan	China India Japan Republic of Korea	China India Japan Republic of Korea
America	Canada USA	Argentina Brazil Canada USA	Argentina Brazil Canada Chile USA	Brazil USA	Canada USA
Europe	Armenia Australia Bulgaria Estonia France Hungary Italy Kazakhstan Lithuania Poland Romania Russian Federation Slovenia Ukraine United Kingdom	France Russian Federation Switzerland United Kingdom EURATOM (EC)	Armenia Belarus Belgium Bulgaria Czech Republic France Germany Italy Kazakhstan Netherlands Russian Federation Slovakia Spain Switzerland Turkey Ukraine EURATOM (EC)	Belarus France Germany Italy Kazakhstan Russian Federation Switzerland United Kingdom EURATOM (EC) ISTC OECD/NEA	Belgium France Germany Russian Federation Sweden Switzerland United Kingdom
Africa	Ghana Jordan Morocco Oman Senegal	South Africa	Algeria Morocco South Africa		South Africa

FIG. 2. International cooperation on the FR cycle.

implementing methods for the assessment of Generation IV systems against GIF goals in the fields of economics, proliferation resistance and physical protection, and risk and safety, and a Senior Industry Advisory Panel has been set up to provide advice on GIF R&D priorities and strategies to help prepare for future commercialization of Generation IV systems. GIF does not concern the fuel cycle. The road map was created in 2002 and currently cooperation is being promoted by Canada, China, France, Japan, the Republic of Korea, the USA, South Africa and Switzerland, and the European Atomic Energy Community (Euratom) which ratified the Framework Agreement [26].

GIF is a forum where its member countries and Euratom that have FR system technologies bring together their R&D results, or alternatively, carry out joint research. The development scope of GIF is a preliminary step for actual construction of facilities. The development goal of GIF includes four items: (i) sustainability (long term and efficient resource use, minimization of waste, etc.); (ii) economics, safety and reliability; (iii) proliferation resistance and (iv) physical protection.

To date, the system agreement that provides modality and member's rights, etc., for actively promoting cooperation has been approved for the SFR, GFR and supercritical water reactor systems. Four projects have begun and one project is under preparation for the SFR, for which cooperation is the most advanced. Of the five projects, it is the Global Actinide Cycle International Demonstration project that typically shows the cooperation of GIF to develop technologies practically. In this project, irradiation tests of minor-actinide-bearing fuel is to be carried out through USA–France–Japan trilateral cooperation. Essentially, the USA offers the minor actinide raw material, France processes this to produce the test fuel pin for irradiation and Japan conducts the irradiation and the post-irradiation tests. Further, the three countries jointly execute the physical property measurement, the evaluation of irradiation behaviour and the deliberation of long term test plans.

3.2. INPRO

INPRO was established in 2001 and based on a resolution of the IAEA General Conference (GC(44)/RES/21). The main objectives of INPRO [27] are to:

(a) Help ensure that nuclear energy is available to contribute, in a sustainable manner, to meeting the energy needs of the 21st century;
(b) Bring together technology holders and users so that they can consider jointly the international and national actions required for achieving desired innovations in nuclear reactors and fuel cycles.

INPRO's current membership consists of 30 IAEA Member States plus the European Commission. Membership is expanding all over the world, reflecting the cooperation that technology user countries can profit from.

Up to now, INPRO has executed six tasks. Task 1 (INPRO Methodology) was developed first. In INPRO Methodology, the requirements of the Innovative Nuclear System (INS, including reactor and cycle) consistent with the sustainable development of a given country, are divided into three stages to achieve INPRO's first objective: (i) basic principle, (ii) user requirement and (iii) criteria which are applied to seven fields such as economics, safety, environment, waste management, proliferation resistance, physical protection and infrastructure (nation, region and international infrastructure). By using the Methodology with its manual, the members can objectively evaluate whether the nuclear power systems, the countries and the development scenarios which are being developed domestically or which are being introduced from other countries are able to contribute to sustainable development. Besides this, there exists Task 2 (evaluating INS by using the developed INPRO Methodology and feeding back the result to improve the Methodology), Task 3 (examining the reference scenario), Task 4 (examining the infrastructure based on the law and the system for introducing the transportable nuclear installations that can be used for developing countries and niche applications), Task 5 (promoting dialogue among the technology holder countries and the technology user countries, and Task 6 (twelve projects that the members are jointly promoting from the viewpoint of safety and non-proliferation, etc.). From 2010, these tasks are scheduled to be re-organized into five programmes: (i) Nuclear Energy System Assessment (NESA), (ii) Global Vision, (iii) Innovations in Nuclear Technology, (iv) Innovations of Institutional Arrangements and (v) INPRO Dialogue Forum [28].

In 2005–2007, the Joint Study on an INS based on Closed Fuel Cycles with FRs was executed as part of the application research that used INPRO Methodology with the participation of Canada, China, France, India, Japan, the Republic of Korea, the Russian Federation and Ukraine [29]. The purpose of this joint study was to demonstrate the potential of adaptability and clarify the characteristics of the FR cycle system that can supply energy in a sustainable manner in the 21st century through an international evaluation using the INPRO Methdology.

The main activity of INPRO is to evaluate the INS. Additionally, it has a mechanism (Joint Initiative) to facilitate project coordination by the IAEA, where the members form groups and procure the capital, personnel, materials and equipment.

3.3. GNEP

The United States Department of Energy announced the GNEP plan in 2006. GNEP aims to balance competing goals for the expansion of use of nuclear energy worldwide and nuclear non-proliferation, and advocates the execution of seven items, including the demonstration of recycle technology with high nuclear proliferation resistance and the development of an advanced burner reactor [30]. In August 2006, the schedule for the two stages (the two-track approach) was shown from the viewpoint of urgent issues such as processing spent fuel and utilizing experiences from industry. This approach was set up to divide the entire development plan into track 1 and track 2 and aimed at achieving complete development by around 2020. Track 1 is to develop a fuel cycle facility that processes spent fuel from LWRs and demonstration FRs using existing technologies as much as possible and to recruite the technical expertise and receive input from domestic and overseas industries. Track 2 aims at the execution of research that uses advanced cycle technology and the construction of an advanced fuel cycle facility which reprocesses spent fuel from FRs and fabricates minor-actinide-bearing fuel.

The first ministerial level meeting was held in May 2007 with the participation of representatives from China, France, Japan, the Russian Federation and the USA. In September 2007, sixteen countries signed the Statement of Principle at the second ministerial level meeting and an international cooperative relationship for GNEP was established. R&D on the advanced nuclear fuel cycle is continuously promoted as the AFCI through R&D activities, as GNEP was stopped in the USA with the election of President Obama. Moreover, it was agreed at the ministerial level meeting in October 2009 that international cooperation on nuclear energy infrastructure, nuclear fuel supply guarantee, spent fuel management service, etc., would be addressed [31].

3.4. TWG-FR and TWG-NFCO

The TWG-FR is a technological working group set up within the IAEA in 1968 to support R&D of FRs in its signatory countries. The working group has been playing a role in promoting international cooperation towards the commercialization of FRs, including next generation reactors and accelerator driven systems, by implementing activities such as collaborative R&D projects from design to construction, operation and decommissioning, and facilitating information exchange with respect to various scientific and technological topics. The TWG-FR is offering a place for international cooperation, reviewing the R&D status and results in each country regularly and submitting recommendations to the IAEA, etc. Currently, fourteen IAEA Member States and three international organizations participate in the TWG-FR [32].

The TWG-NFCO provides the IAEA with guidance for its programme on spent fuel management and advanced fuel cycles, including advanced recycle technology and FR fuel cycles. It started in 2002 by integrating the former Regular Advisory Group on Spent Fuel Management (launched in 1984) and the International Working Group on Nuclear Fuel Cycle Options (launched in 1997). Typically, fourteen IAEA Member States participate in this working group [32].

3.5. International cooperation in the future

The current state of global international cooperation for the FR cycle has already been referred to. As regards future international cooperation, it is also important to look at international cooperation regionally, where culture and ideas are similar, and therefore, the influence of distribution, etc., is largely expected. Concerning nuclear energy development, advanced, developing and interested countries (particularly nuclear sensitive countries) are coexistent in Asia and the Pacific Basin. Therefore, political and interdisciplinary mutual collaboration is necessary to promote sound nuclear energy development. Although the Forum for Nuclear Cooperation in Asia and the Pacific Basin Nuclear Conference have already been held, it seems necessary to expand the number of participating countries and the fields of cooperation. Moreover, it is required that energy resources be secured and the approach taken for the abolition of nuclear weapons be strengthened by regional management of nuclear material for peaceful use (guarantee of supply and takeover). In any case, the role of Japan, which is one of the most advanced countries with nuclear technologies and the only nation to have been bombed with atomic weapons, is very important.

4. NATIONAL TRENDS AND STRENGTHENING OF INTERNATIONAL COOPERATION

4.1. National development trends and common issues

The status of energy policy for FRs in each country differs according to its circumstances. The countries mentioned in Section 2, for the most part, recognize the central role of the FR as a future source of power. In India, on the other hand, the FR is placed at an intermediate stage in the realization of the Th fuel cycle system, although the US emphasis is placed on the means to achieve non-proliferation. Therefore, the plan for achieving the commercialization or the target of the FR cycle should match each country's demand. Moreover, in a technical aspect, there are many kinds of fuel and fuel cycle technologies and many different forms of reactors as well as fuel cycle technologies. In this respect, it is possible that various technologies can 'coexist' with future FR cycle systems as well as with thermal reactor systems, which comprise various technologies such as LWRs, HWRs and graphite moderated reactors. However, FRs are inevitable for future global energy security.

On the other hand, safety and non-proliferation are common demands made on nuclear energy.

4.1.1. Safety

Accidents that occur in nuclear power plants and fuel cycle facilities have the potential to damage extremely wide areas over the long term. In addition, when an accident occurs, it might not be solved as a technical problem, but require political action, and its influence might spread not only across the country but also globally. Originally, the purpose behind the development of the FR cycle system was to maintain a steady energy supply over the long term. If a problem such as safety is raised, it is feared that the energy supply might be discontinued over the long term. Because the safety of nuclear power is globally recognized as a high priority issue, it is necessary to create a global consensus regarding handling the re-criticality issue in a hypothetical core disruptive accident in FRs, etc. It is necessary to establish a safe handling technology for sodium so that neither accidents nor incidents such as sodium leakage and sodium–water reactions occur in SFRs.

4.1.2. Non-proliferation

The FR cycle contains a quantity of Pu in the system. Therefore, it is necessary to take measures not to isolate pure Pu in any stages of the system. The FR cycle can contribute to energy security when it is put to practical use, and many countries may be expected to introduce the FR cycle in the future. Some countries that develop FRs examine the small, long lived FR that can be used in developing countries and in those with poor electricity infrastructure. Considering the characteristics of the FR cycle system and the possibility of deployment worldwide, it is clear that attention to non-proliferation should be required more than ever. Assuming that the FR cycle system is deployed worldwide, an international consensus should be fostered concerning the development of an applicable and reasonable safeguards system and the techniques for evaluating nuclear non-proliferation, etc. Moreover, strengthening the authority and activity of the IAEA, fully supporting President Obama's remarks on nuclear abolition and promoting bilateral or multilateral dialogues with consideration of nuclear geopolitics are also required.

As for the above-mentioned two points, safety and non-proliferation, it is necessary to reach a solution based on a globally common consensus for any FR cycle systems.

4.2. Strengthening of international cooperation

International cooperation and coalition are indispensable for each country to introduce and establish FR cycle technology in the future. International R&D cooperation, collaboration and joint development could not only reduce R&D risks and costs but also provide the opportunity to produce global standard technologies, while separating areas of competition from those of cooperation. Further, it is likely to become a driving force for global peace and regional community construction. When international cooperation is undertaken, peaceful use and safety of nuclear energy become major premises, and securing nuclear non-proliferation, safeguards and nuclear security should be indispensable, and therefore, that the countries concerned are willing to share development goals of these ideas.

In the above-mentioned multilateral cooperation, each framework seeks to advance activities and produce a result. On the other hand, scope and national organization involved in each activity are limited in the framework. For instance, GIF is a forum to develop actual technologies of the next generation nuclear energy systems and currently it advances international cooperation on six reactor systems towards commercialization. The participants are limited to the countries and international organizations that have the technologies, and the result is only shared internally. Moreover, the actual construction phase and the fuel cycle project have been excluded from international cooperation. It is possible to participate in INPRO as long as the IAEA Member States and regions include user countries which plan for nuclear power generation in the future. INPRO is a forum in which members consider technology and institution, and has several differences from GIF in terms of the member composition, the active areas such as evaluation approach, institution and technological development, and the viewpoint of activity which values the expectations for the INS from the user countries. GNEP aimed at the development of reactor and fuel cycle technology for a certain period of time, but it has returned to its first purpose of contributing to the deployment of nuclear power worldwide and ensuring that non-proliferation activities are still continuing.

In the light of the purpose of international cooperation that R&D should be efficiently advanced, it is important to supplement activities in each country without overlapping each other. As for GIF and INPRO, they supplement each others' activities and the specialists of both frameworks have even participated in the other party's working group for safety, nuclear non-proliferation, etc., since inauguration. The representatives of GIF and INPRO participate in the other party's decision making meetings, GIF policy group meetings and INPRO steering committee, as observers, and both parties communicate to avoid the repetition of their activities. Moreover, GIF and the IAEA send observers to the GNEP ministerial level meetings.

If the direction of these activities can be brought together, it is expected that the timeframe towards commercialization of the FR and the time needed to make it acceptable worldwide can be shortened. Each country should deliberate how to build these frameworks into its national development strategy while recognizing its features and the limits of the frameworks. Also, it is necessary to promote development domestically and globally.

In addition, it is necessary to publish the results of development actively and promote the exchange of information by using international meetings such as the International Conference on Fast Reactors and Related Fuel Cycles (FR09), TWG-FR (these meetings focus on the FR), International Congress on Advances in Nuclear Power Plants (ICAPP, focusing on advanced reactors), GLOBAL, TWG-NFCO, an OECD/NEA workshop for separation and transmutation, ACTINET/J-ACTINET, PBNC, International Forum of Nuclear Non-proliferation (these meetings focus on fuel cycle technology). Moreover, it is scheduled to hold an international conference for advanced reactor and fuel cycle technology, where the Asian Nuclear Prospect (ANUP) will be held alternately with GLOBAL in Asia, where nuclear power is increasing significantly. The first meeting was held in Kobe in 2008 and the second (in India) and the third (in China) are scheduled to be held in 2010 and 2012, respectively [33, 34]. The Japan Atomic Energy Society is now preparing to establish a division for advanced reactors in which activities such as R&D, international activity, exchange/education of researchers are planned with regard to Generation IV reactors, future nuclear energy systems and related fuel cycle technologies.

4.3. Human resource development

FR R&D should be conducted over the medium to long term. Therefore, human resource development and knowledge preservation should be the most important issues in each country. These should be performed by international collaboration, collaboration among research institutes, industries and universities, and public involvement. In these international collaborations, the IAEA should play an important role. This international collaboration for FR R&D also contributes to strengthening the non-proliferation of FR.

5. CONCLUSION

The development of the FR cycle adjusted to fuel cycle technology is indispensable and the steady development of both is expected so that the FR may function efficiently. The countries that possess nuclear energy technology and that undertake advanced R&D of FR cycle technology invest the resources of personnel, finance and facilities according to the national developmental strategy. Some countries achieve operating a particular scale FR. However, more investments in R&D are necessary to take FRs to a commercial level. Moreover, safety is a major premise in the use of FR cycle technology, and a countermeasure for a non-proliferation issue due to deployment should not be decided by one country but requires a multilateral consensus. It is thought that international cooperation has become more important for developing FR cycle technology and the related technology.

More specifically, for the development and utilization of FR cycle systems with respect to each country's concept, it is essential to organize the technologies and concepts which should be commonly recognized worldwide and build a framework to make them standardized. The existing frameworks such as GIF and INPRO are considered effective in helping to achieve this. Furthermore, vigorous promotion of international cooperative developments enables the formation of an international consensus on major technologies for the FR cycle, as well as the saving of resources by infrastructure sharing. Information exchange through international conferences such as this one (FR09) plays an important role in achieving this goal. Continuously holding international conferences such as FR, GLOBAL and ANUP, is expected in the future. Further, it is also necessary to promote the vigorous exchange and transmission of information through academic conferences, nuclear energy seminars and exchanging opinions with the media and politicians, etc.

Finally, it is firmly confirmed that Japan will make a great effort to facilitate global cooperation as it is one of the few nations to have both experimental and prototype FRs. Further, Japan, as a non-nuclear-weapon nation, considers that it should play an important role in studying and creating a system where peaceful FR cycle technologies can be developed and would like to contribute actively to the international community. Through such efforts, Japan seeks to contribute to the establishment of an FR cycle technology that coexists with nuclear non-proliferation.

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RETROSPECTIVES AND ADVANCED SIMULATION

(Plenary Session 6)

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FRENCH SFR OPERATING EXPERIENCE

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Abstract

During the last twenty years, the most significant events that occurred at the two French sodium cooled fast neutron power reactors (SFRs), Phénix and Superphénix (Creys-Malville), were:

(a) The four negative reactivity transients which occurred at the Phénix reactor and the many studies that were carried out to explain these phenomena and demonstrate the safety of the reactor (1989–1993);
(b) The modification of the Creys-Malville secondary circuits and their environment for protecting the reactor against large sodium spray fires (1992–1994);
(c) The French Government decision to shut down definitively Superphénix in 1997;
(d) The lifetime extension of the Phénix reactor, which needed a number of modifications and inspections, in accordance with the updated safety requirements (1998–2003);
(e) The beginning of the decommissioning of the Creys-Malville plant, which needed the development of specific processes and methods;
(f) The reliable operation of the Phénix plant during its last irradiation cycles (2003–2009) and the completion of the experimental programme on nuclear waste transmutation;
(g) The final test campaign on the Phénix reactor (2009).

Operating feedback from the Phénix and Superphénix reactors and incidents occurring in them are analysed in order to learn lessons for the design and operation of future SFRs. Despite a contrasting operating history for the two reactors, particularly in view of a different political context, the lessons learned from incidents occurring and from recorded scientific and technological knowledge are consistent. In terms of the core and fuel, controlling reactors, components, handling subassemblies and components, materials and sodium technology, maintenance and inspection, France currently has a sizeable cognitive knowledge base. This provides a sound starting point to achieve the current requirements for risk prevention (technical, human and financial), and cost and availability targets to be applied to future SFRs. Feedback from their operation and incidents that have occurred are analysed to learn lessons for the design and operation of future SFRs. Efforts are also made to distinguish between the areas in which a sound knowledge base has been acquired, those needing improvement due to

technological progress inside and outside the nuclear sector and, lastly, those requiring more significant innovation, possibly breaking with previous practices, to achieve the current requirements for risk prevention (technical, human and financial), and cost and availability targets. Following a reminder of the most significant events of the last 20 years regarding the French SFRs, this paper presents the acquired scientific and technological knowledge. Then, potential lessons to be learned for future SFRs, which were developed from this operating feedback, are suggested in terms of their prospective use.

1. PHÉNIX: THE LAST 20 YEARS OF OPERATION

The Phénix[1] power plant experienced four negative reactivity trips in August and September 1989 and September 1990 caused by a very rapid and high amplitude variation in the signal from the power range neutron chambers. The initial explanations put forward related to interference in the measuring chains, which had been modified during the ten-yearly outage in 1989, without these chains being shown to be particularly sensitive, however. The elements collected after the third event in summer 1989 led to the power change being attributed to a volume of gas moving through the core. This explanation seems consistent with both the observation of cover gas overpressure indicated and possible plugging of the diagrid trap subassemblies. It was decided to shut down the power plant whilst awaiting the collection of additional information.

Having analysed this scenario and its consequences, and taken preventive measures, the reactor restart was authorized in December 1989. Two irradiation cycles took place. This explanation nevertheless was negated by the occurrence of a fourth event in 1990. The CEA therefore instigated a major investigation programme. An expert assessment committee was created to coordinate all the investigations covering many specialities (neutronics, hydraulics, mechanics, chemistry, etc.).

The expert assessment culminated in 1991 with the following two conclusions. The event initiating the anomalies was not clearly identified; however, it was strongly supposed that the variation in reactivity was caused by radial expansion of subassemblies, then a return towards the centre (core flowering). In addition, the safety analysis based on different phenomena showed that these

[1] In France, two sodium cooled fast reactors (SFRs) have been operated: Phénix (563 MW(th)–250 MW(e)) from 1973 to 2009 by the Commissariat à l’énergie atomique (CEA) and EDF combined (80% and 20%, respectively) and Superphénix (3000 MW(th)–1200 MW(e)) from 1985 to 1998 by NERSA made up of EDF (51%), ENEL (33%), RWE (11%) and other electricity producers from Belgium, the Netherlands and the United Kingdom.

events did not indicate disorders in the reactor's internal structures, especially core support structures, and that, regardless of the initiating mechanism, this raised no doubts over reactor safety. The same conclusion was reached for the Superphénix reactor.

Having boosted the measuring and recording resources for reactor monitoring, a ten day phase of power operation was authorized in February 1993. The goal was to check the behaviour of the reactor, especially the core, test the instrumentation and investigate certain scenarios. The absence of anomaly in the reactor was confirmed, but no decisive factor in identifying the phenomenon emerged. After a campaign to repair the secondary sodium systems and replace the intermediate heat exchangers, the reactor started operating again in December 1994.

At the same time, a project to extend the service life of the Phénix reactor was undertaken to perform experimental irradiations in the transmutation of long lived radioactive waste. Initially designed for a ten year period, the CEA finally set the additional irradiation period at six cycles of 120 equivalent full power days (EFPD), which corresponded to about six years of operation. The first renovation works started in 1994; the Phénix power plant performed two operating cycles in 1995 and 1998 and the most significant work took place from 1999 to 2002 [1].

The studies and modifications designed to ensure and improve the safety of the facility mainly focused on [2]:

(a) Lowering the average temperature of the hot pool from 560 to 530°C to reduce fatigue–creep damage, which is difficult to model, and ageing of equipment, particularly titanium stabilized steels;
(b) Reducing the maximum reactor power to 350 MW(th) to guarantee evacuation of the residual power (initially underestimated) should the intermediate systems fail (one intermediate circuit was finally shut down);
(c) Ultrasonic examination (inside the sodium maintained at 150°C) of welds on the conical shell that supports the core and connects it to the main vessel;
(d) Televisual inspection of the core cover plug, the upper internal structures of the reactor block and the geometry of the subassembly network, having drained half (400 t) of the primary sodium into the storage tanks;
(e) Introducing an additional shutdown rod similar to those installed in Superphénix, effective even if subassemblies are seriously deformed;
(f) Replacing portions of intermediate circuits made of 321 steel (sensitive to delayed reheat cracking of welds in the complex geometry areas and subject to high temperatures), including in the steam generators;
(g) Strengthening the earthquake resistance of all buildings, particularly steam generators and pressurized water and steam piping (hazard risk in neighbouring sodium systems);

(h) Modifying the emergency cooling system to ensure its operation in the event of an earthquake or a major sodium fire on the site;
(i) Partitioning of areas containing pipework carrying sodium from areas containing pressurized water or steam pipework and also partitioning of the two intermediate sodium systems which remain in use.

The Phénix power plant carried out its last irradiation cycles from 2003 onwards at the rate of the insertion and removal of experimental subassemblies and outages, during which the number of inspections and the maintenance increased dramatically. A fifth sodium–water reaction in a reheater module in September 2003 (the leak was probably caused by an initial manufacturing fault) and a sodium leak in the main piping of an intermediate system in August 2007 (through a weld repaired in 1997) were the only significant events to disturb this period. At the beginning of March 2009, the power plant finished its power phase operation. ‘Final’ tests subsequently took place in 2009 to acquire special neutron, thermohydraulic and safety results before final offloading of core subassemblies and the deconstruction of the facility. These tests are described in another paper given at this conference [3].

The outcome of the Phénix power plant operation over 35 years [2] is set quantitatively at 4580 EFPD, 128 500 h of connection to the electrical grid and 28 billion kW·h of electricity generated. The capacity factor for the power plant (ratio between the gross electrical energy produced and the product of the rated capacity over time) is equal to 32%. Compared with the authorized powers and having deducted outages to analyse negative reactivity trips and renovated the plant, the capacity factor is 54%. The plant’s availability rate, excluding scheduled outages, is in the order of 75%, with the main unavailability being due to the intermediate heat exchangers and steam generators as well as the electricity generation facility. The record for uninterrupted operation connected to the grid is 151 d (October 2006 to March 2007).

2. SUPERPHÉNIX: THE LAST 20 YEARS OF OPERATION

In June 1992, the Creys-Malville plant was technically ready to restart after having ‘answered’ to the Ministers’ requirements about the operating conditions [4]. The Nuclear Safety Authority considered that “the startup of Creys-Malville can, in terms of safety, be authorized under a set of limitations and precautions” relating particularly to the risk of spray sodium fires (after the fire in the Almeria solar power plant in 1986, it proved that this type of sodium fire could have more significant consequences in terms of pressure and temperature than the pool fires considered during the design phase). However, on 29 June 1992, the Prime

Minister decided that the relevant work had to be carried out before any startup (also subject to the conclusions of a public enquiry and a report from the Minister for Research on the advantage of the plant with respect to the incineration of radioactive waste).

As soon as the studies were sufficiently advanced, the ‘sodium fire’ work was undertaken around the intermediate circuits, in particular [5]:

(a) Improving sodium leak detection by installing several hundred new ‘sandwich’ detectors around the main welds;
(b) Partitioning rooms into 100 m^3 lots with metal partitions restricting the range of a sodium jet and the amount of available air;
(c) Opening outlets in the reactor building containment to cut back the overpressure and evacuate the hot gases and aerosols produced during a major fire, fitted with quick opening check valves capable of subsequently closing and providing a seal;
(d) Applying insulation to the concrete walls to prevent moisture being released during a sodium fire which could, in some conditions, react with the sodium and release hydrogen.

Work ended in the first half of 1994. The administrative procedure finally culminated in a new ‘authorization decree’ (July 1994) and the ministerial authorization for criticality. The startup was deliberately slow and gradual to allow the operating teams to get a ‘feel’ for the facility; virtually all the tests that would have been performed on a new core were carried out, as were requalification tests for modifications at various power levels.

During the second half of 1994, the plant operator detected and monitored a leak of argon supplying the sealing system in an intermediate heat exchanger. The reactor was shut down at the end of December and investigations confirmed the presence of a crack about 30 mm long in the feed tube. A repair solution was developed as an alternative to replacing the defective heat exchanger with a spare, which would necessitate special heavy handling and dome rotation. The repair was implemented in a few hours in July 1995; a sleeve was inserted into the tube (diameter: 22 mm) over about 12 m. It was positioned at the crack, then expanded by pressurizing with water to seal the assembly.

Startup via successive series then continued. The reactor reached the maximum power authorized by the Nuclear Safety Authority (90% of nominal power) in October 1996. The outage scheduled for the first half of 1997 was intended to reduce the breeding ratio of the core by replacing fertile subassemblies with steel subassemblies. Three experimental subassemblies (destined to demonstrate the capability of the reactor to ‘incinerate’ plutonium and neptunium) had to be inserted into the core. Advantage was also taken of the outage for an initial inspection of tubes in a steam generator with an ultrasonic technique.

These activities took place normally, despite a warning 'shot' from the Council of State order dated 28 February 1997 cancelling the decree of 11 July 1994. The second shot came from the new Prime Minister, who announced on 19 June that "the fast breeder reactor known as Superphénix will be abandoned" [6, 7]. Preparations for the final shutdown of the plant started in October 1997. The technical and safety studies put together by the plant operator and their processing by the Nuclear Safety Authority culminated in the signing, on 30 December 1998, of the authorization decree for the first stage of the final shutdown of the reactor. On 30 March 2006, two additional decrees authorized the complete dismantling of the reactor (treating the sodium, dismantling the reactor block and demolishing buildings) as well as extending the missions of the Fuel Assembly Storage Facility (APEC).

The first operations place the equipment and systems no longer of use out of service once and for all before dismantling them (external electric lines, balance of plant, some emergency diesel generators, reactor shields, outside stacks, water and steam lines, etc.). The core subassemblies are unloaded (fertile and fissile), washed of their residual sodium and transferred to the pool in the APEC. The control rods and side neutron shielding follow the same route but are stored in shipping casks on the APEC premises. Small components (control rod mechanisms, cladding rupture detection modules, etc.) are removed from the reactor, washed, cut up and conditioned in waste containers. The large components (primary pumps and intermediate heat exchangers) have been suffering the same fate from 2009 onwards. A sodium treatment facility (conversion into soda and incorporation into concrete) was installed in the former turbine hall and a building was being constructed nearby to store the fabricated concrete blocks for about twenty years [8]. The facility started operating in 2009 and should continue for five years. On-site sodium risk will have been eradicated at the end of this stage. Lastly, the dismantling studies for the reactor block have been refined. The option chosen is carbonation of the residual sodium contained in the main reactor vessel, after draining and cutting up the reactor block under water using remotely handled tools. This deconstruction stage should last about ten years, at which time the buildings will be demolished (except for the APEC buildings).

The outcome of the operation of the Creys-Malville power plant during these 11 years is set quantitatively at 319 EFPD, 11 000 and 12 300 of grid connected hours for each of the two turbogenerators and 8.3 billion kW·h of electricity generated. The capacity factor for the power plant (ratio between the gross electrical energy produced and the product of the rated capacity over time) is equal to 6.3%. With respect to the powers authorized by the Nuclear Safety

Authority and the periods deducted during which the Nuclear Safety Authority revoked the plant's operating licence, the capacity factor is 21%. Lastly, if the production periods only are taken into consideration, the capacity factor is equal to 41.5%, which is representative of the difficulties in maintaining the plant in operation; still very far from industrial operation. Nevertheless, during the last year in operation, the power plant's availability rate, excluding scheduled outages, was in the order of 95%.

3. ACQUIRED SCIENTIFIC AND TECHNOLOGICAL KNOWLEDGE [9]

The fuel used in France in the form of mixed uranium and plutonium oxide has provided excellent operating feedback. Advances in cladding materials have, little by little, enabled burnup to be increased. This has increased from 50 000 MW·d/t for the Phénix startup to 90 000 and 115 000 MW·d/t (in the centre and at the edge of the core, respectively) in the middle of the 1980s (i.e. a damage rate of about 100 dpa). Experimental subassemblies reached far higher specific burnup values. The record belongs to the BOITIX 9 subassembly (hexagonal tube in EM 10 steel and cladding in cold worked 15.15 Ti steel), which accumulated 144 100 MW·d/t (156 dpa) in Phénix.

This significant gain was achieved by keeping the number of clad failures to a minimum. Of some 150 000 fuel pins irradiated in Phénix during its 35 years in operation, there were only fifteen 'open' cladding failures (none at Creys-Malville), including half in experimental pins irradiated beyond standard characteristics. The qualification and use of titanium stabilized steels (316 Ti and 15.15 Ti) for the cladding, hexagonal tubes and structures in fuel subassemblies has confirmed the considerable knowledge acquired on the essential phenomena of swelling in steels under a major neutron flux.

Thanks to the reprocessing of irradiated pins, the fuel cycle was closed several times (i.e. the plutonium recovered in the reprocessing workshop was reinserted into new subassemblies and irradiated once again in Phénix). A significant breeding rate was thus demonstrated industrially: 16% of plutonium was produced in addition to the equivalent of the initial quantity.

The flexibility of the Phénix reactor played a major part in the gradual introduction of increasingly varied and numerous experimental irradiation devices. As a result of its high neutron flux, the presence of targets had little influence on maintaining the chain reaction and the ease of loading and unloading for isolated subassemblies, the Phénix power plant was transformed into an attractive experimental tool, especially for destroying or transforming undesirable radioactive elements whilst continuing to generate electricity.

Many factors make it particularly easy to operate an SFR:

(a) No pressurization of the primary coolant;
(b) High thermal inertia of primary cooling and intermediate systems;
(c) Control of power by a single control rod position;
(d) No xenon effect;
(e) No soluble neutron poison (such as boron for the water reactors).

The fact that there is no problem or difficulty during reactor operation deserves special mention. However, there is core sensitivity to changes in reactivity during assembly movements, highlighted in particular by studies following the negative reactivity trips in Phénix. Core compaction is impossible due to the contact of subassemblies at the pads on their heads. On the other hand, subassembly movement towards the outside of the core is mechanically possible (no core restraint) and produces sufficiently important changes in negative reactivity to make measuring and regulating components react or even cause automatic shutdown.

The Phénix reactor has proved its ready availability, setting aside the years 1990–2003 devoted to reassessing its safety and the corresponding renovation work. The same applied to the Superphénix reactor during periods in which power operation was authorized, notwithstanding the fine tuning difficulties during successive startups. More generally, the operators have found these reactors easy to run (the conventional electricity generating utility, responsible for one third of unscheduled shutdowns in Phénix and two thirds in Creys-Malville, frequently caused them more problems). The particularly high performance expected in terms of thermal efficiency (over 40%) was achieved. It is also clear that the facilities produce few effluents and little radioactive waste.

The Phénix plant was also capable of operating at reduced load (with only two out of three secondary sodium systems in operation). This meant that in the early days, it could continue to operate during rotating repairs to intermediate heat exchangers or steam generators and, at the end of its life, optimize the renovation work required to satisfy the new safety requirements without reducing the performance of experimental irradiations in the core.

The operating experience related to the main components is mixed. Curiously, the operation of reactor coolant pumps (active equipment) was excellent, whereas the intermediate heat exchangers (static equipment) had to be repaired several times (design defects, corrected in the Creys-Malville intermediate heat exchangers, or manufacturing faults).

The steam generators in the Phénix power plant suffered five sodium–water reactions. The first four were due to the combination of a design fault and an inappropriate operating procedure, resulting in thermal shocks and mechanical fatigue, and ultimately, fatigue cracking. The first was complicated by too much time being taken to detect, drain and inert the steam generators: 30 kg of water reacted with the sodium. The subsequent improvements limited the quantities of water involved to 1–4 kg in the following reactions. The fifth sodium–water reaction, in 2003, was due to a manufacturing fault.

One potentially aggravating factor in sodium–water reactions is the effect of the pressurized steam jet that reacts with the sodium and creates a sort of 'torch' where the jet can rapidly pierce either a neighbouring tube or the shell of a steam generator (wastage effect).

The lessons drawn from analysing these incidents were implemented in the design and operation of Creys-Malville:

(a) Reliability and speed in detecting the sodium–water reaction (hydrogen, acoustic);
(b) Automatic shutdown accompanied by rapid depressurization of the water and steam systems, then nitrogen injection in the affected steam generator;
(c) Rupture disks to limit the pressure increase in the intermediate system;
(d) Integrity of the sodium envelope in the steam generator and, more generally, the intermediate circuit, in relation to the most violent sodium–water reactions;
(e) Choice of suitable materials and quality of fabrications.

It should also be noted that the expert assessment and replacement of defective modules were systematically possible due to the modular steam generator design in the Phénix power plant. If a sodium–water reaction had occurred in the Superphénix reactor, the defective tube, and any tubes potentially suffering side damage, could have been plugged but could not have been extracted easily for expert assessment.

Handling subassemblies in an SFR is very different from the known procedure in water reactors. First and foremost, the opacity of the sodium forces work to be carried out 'blind' as long as the subassemblies are in the reactor or in the external sodium storage tank. Methods to control movements and check that there are no obstacles have been developed ('vision', particularly by ultrasound) to offset this disadvantage. The remaining sodium then has to be removed from the subassemblies so that they can be stored temporarily under water. These operations require biological shields and are performed remotely. In the Phénix plant, a series of additional operations involves cutting spent subassemblies and placing them in suitable leaktight canisters, firstly, the fuel or fertile pins, and secondly, the steel structures from the subassemblies, for temporary storage in other CEA facilities.

Overall, handling operations in Phénix have provided positive feedback. Basically, it should be noted that core renewal campaigns took gradually longer, firstly due to the ageing of equipment (most frequent breakdowns) and secondly to the more stringent procedures for monitoring subassembly movements, imposing more controls and hold points during operations.

At Creys-Malville, the only significant operation involved the final unloading of all the subassemblies in the core. This operation suffered due to no decoupling area for activities between the reactor and the pool (no external storage drum), the context of final reactor shutdown, the fact that the facilities were used for the first time fifteen years after their installation and the existence of highly restrictive safety criteria (residual sodium mass).

The behaviour of equipment and materials provides as much information when all goes well as when contingencies occur, which although frequently providing a wealth of new information, are unfortunately detrimental to the 'healthy' operation of the power plant. The materials used, especially the stainless steel in the primary cooling system, have proved to be perfectly suitable for their use, with two exceptions (see below).

The sundry equipment, especially designed for SFRs, has, overall, been proven to perform satisfactorily: electromagnetic pumps, plugging indicators and cold traps, thermocouples, and equipment for cladding failure detection and location, sodium leak detection, hydrogen detection (sodium–water reaction), etc. The assessment is more mixed in terms of behaviour, for some equipment has set off alarms inadvertently, suffered malfunctions, needed repairing and even caused production losses. The fact that these defects were more frequent at Creys-Malville than in Phénix raises the question of reactor size optimization, and more importantly, reactor design and the quality of fabrications.

The leaks from the intermediate heat exchangers in Phénix have underlined the potential effects of mixing sodium flux at different temperatures (causing differential expansion of shells). They have also demonstrated the possibility of extracting, cleaning, decontaminating, repairing and recommissioning major components, which have operated for several years with sodium and under irradiation. The sodium leak in auxiliary pipework at Creys-Malville in May 1990 had the same cause (in this instance thermal striping was the cracking mechanism).

The minor sodium leak in Phénix in May 1986 revealed a particular phenomenon, confirmed and analysed subsequently by several CEA tests and, more recently, by the sodium leak detected in August 2007. Following the leak, the sodium came into contact with the insulation, triggering a chemical reaction (exothermal creation of sodium hydroxide). Caustic corrosion of the piping steel followed in the form of a groove at the limit of the sodium plume. This risk has been taken into account by limiting the maximum time between the appearance of a sodium leak detection alarm and the draining of the affected circuit (normally causing reactor shutdown), if in the meantime it has been impossible to refute it by in situ inspections. This problem was also the reason for developing a leak before break approach: frequent inspections of weld states and rigorous leak detection ensure that, if an earthquake occurs, it does not encounter major pipework made vulnerable by defects and liable to open under tremors.

The use of 15Mo3 steel when sodium is present has now been abandoned following the cracking of the storage drum at the Creys-Malville power plant. Cracking of 321 steel in the hottest sections of secondary cooling systems has been highlighted by the specific development of ultrasonic inspections of welds. This involves cracking by thermal stress relief of the affected areas during welding, which is highly dependent on operating conditions.

It is also worth noting the problems raised by the design of the core cover plug, which, in particular, is subjected to high sodium temperatures at the core outlet and severe cold shocks during automatic shutdowns. Certain areas of this component in Superphénix had to undergo advanced calculations to justify the acceptability of fatigue–creep phenomena. The safety reassessment for Phénix came up against the demonstration of acceptable damage to core cover plug welds; the visual inspection subsequently showed that these were in good condition.

This operating feedback is fundamental knowledge for designing the various systems and components for future SFRs. It must form the basis for building, paying particular attention to simplifying and industrializing equipment, choosing correctly and qualifying the materials and their implementation processes as well as increasing component reliability and service life.

Lastly, several major maintenance and inspection operations took place in the Phénix reactor and its main systems. It is impossible to produce an exhaustive list here, but notable examples follow.

Replacements and repairs of intermediate heat exchangers, primary pumps and steam generator modules, which were planned in the reactor design, were carried out successfully many times in very short deadlines (e.g. three weeks to replace an intermediate heat exchanger).

Major portions of secondary sodium systems were repaired, with a change in the base metal when 321 steel proved unsuitable for the operating conditions in the hottest parts. On this occasion, an original and effective procedure was set up to weld new portions on to the aged piping in service.

The upper internal structures in the reactor block, especially the core cover plug and the fuel subassembly head network, were examined by remote visual inspection using optical devices inserted into the primary cooling system after draining half the sodium (400 t) in an estimated atmosphere in the order of 100 Gy/h. This inspection showed the excellent state of these structures after thirty years in operation.

The ultrasonic inspection of the conical shell, which supports the diagrid and core inside the main vessel, demonstrated the lack of defects in this structure, which is fundamental to reactor safety, particularly in the case of an earthquake. This used the shell itself as a wave guide from outside the main vessel and over three metres distance at the heart of the primary sodium maintained at 155°C. This operation can be called a world 'first'.

It should be noted that these operations, and more generally, all maintenance work and inspections in the Phénix power plant, took place in vastly superior conditions of radiological cleanliness than found in other types of reactor. In addition, the collective dose received by all the people involved (CEA, EDF, contractors) since the startup of the power plant until its final shutdown is less than 2.5 man Sv (i.e. this is the order of magnitude of annual irradiation of the personnel in two existing EDF pressurized water reactors).

Significant operations also took place in the Superphénix reactor: welds in the main vessel were inspected by the MIR machine, the storage drum was replaced by the fuel transfer chamber, an intermediate heat exchanger was repaired in situ and the tubes in a steam generator were inspected.

4. LESSONS FOR THE FUTURE

The assessment of the Phénix power plant operation is positive, that of the Creys-Malville rather more mixed, but allowance must be made for the technical results and political and media aspects. Consideration should also be given to the early interruption in the Superphénix reactor operation. In addition, the majority of reasons which led France to commit to the Phénix and Superphénix projects forty years ago are topical again: more efficient use of natural uranium resources, eventual risk of tension in the market for this fuel and sustainable development (this expression had yet to be invented) of energy. Added to this is the potential of SFRs to transmute the minor actinides into shorter lived, less harmful products. For this reason, the French Parliament has requested, through the law of 28 June 2006 on radioactive waste management, to state its opinion by 2012 on the advantages of going down this path. An SFR prototype could subsequently be built.

In this perspective, it is essential to analyse impartially the operating feedback acquired to determine the changes required for the future SFRs. This section takes into account the main conclusions to be drawn from the operation of the Phénix and Superphénix reactors, without entering into purely technological lessons learned. As is frequently the case elsewhere, the emphasis will be far more on points for improvement than on those which have already given satisfaction.

The availability of the French SFRs, even apart from shutdowns imposed by safety reassessments and follow-up to incidents such as the storage drum leak or the negative reactivity trips, has not matched that of the existing EDF water reactors. This is partly due to the frequency (Phénix, imposed originally by the desire for fast plutonium production) and the duration (Superphénix, after replacing the storage drum with the fuel transfer chamber) of outages for refuelling. This then is one of the major areas for seeking improvements and even innovations that break with previous choices.

Efforts must also be made in terms of the fuel. Whereas the acquired knowledge provided by the core and the fuel, basically in Phénix, is impressive and rightly deserves national and international recognition, this technology has yet to reach its peak and progress can still be made. R&D is used to improve the characteristics of cores in future SFRs, with longer irradiation cycles (for fuel subassemblies and control rods) and higher burnup as well as sounder safety (void effect, reactivity feedback coefficients, etc.). Core monitoring could also benefit from the far higher performance available with current and future resources.

Malfunctions in the intermediate heat exchangers, steam generators (Phénix only), sodium secondary cooling systems and the installation of electricity generation have also caused significant production losses. As this involves the Phénix power plant, sodium leaks on intermediate heat exchangers and steam generator tube leaks were responsible for production losses equivalent to two years and one year, respectively, even though the reactor design enabled operation at reduced power (two thirds) for a large part of the repair and modification work. Particular attention should therefore be paid to the design of these components through which all the reactor power passes and all available operating feedback should be included.

It is worthy of note that, unlike the ‘popular misconception’, the incidents at the Creys-Malville power plant cannot be directly attributed to the scale jump produced from the Phénix reactor. On the other hand, technical and organizational difficulties are without doubt linked to the size and complexity of the facility. These have also had an impact on core safety characteristics and the means of evacuating residual power, the quality of the construction (factory or on-site fabrications) and the option of using modular (steam generators) or simpler (electromagnetic pumps) components. The choice of power of future SFRs is therefore important.

In addition, the frequency and delay in repairing recorded sodium leaks are clearly not compatible with an equivalent requirement of availability of future SFRs with third generation water reactors. It is therefore particularly important, even without risk for the safety of the facility, to survey the quality of all fabrications (and subsequent modifications) of piping and tanks containing or likely to contain sodium, to reduce this frequency considerably in future SFRs. In addition, their design must allow for immediate detection of a sodium leak (a few minutes), for it to be located promptly (a few hours), assessed by an expert and repaired very quickly (a few days).

Reactors cooled with sodium have characteristics favouring safety, which have been proven during startup tests and operation of the Phénix and Superphénix power plants. They include:

(a) No pressurization of the primary coolant;
(b) Significant margin under normal operation in relation to sodium boiling;
(c) Large thermal inertia of the reactor, which makes it virtually insensitive to the variations undergone by the electricity generating utility;
(d) Leaktightness of the primary cooling system, further strengthened by two safety vessels, and, above all, the option of removing the decay heat from the reactor by natural convection (i.e. in a virtually passive mode).

Overall, the safety level at the Creys-Malville power plant has been identical to that of its contemporary water reactors (according to an assessment by the Nuclear Safety Authority). However, improvements must be made to the design of future reactors for them to achieve the same safety level as third generation reactors: better use of favourable core characteristics, reduced failure probability for decay heat removal, virtual elimination of large sodium fires and violent sodium–water reactions, reduced risk of gas carryover into the primary cooling system, controlled hydrogen production when processing sodium residues, strengthened seismic resistance of structures, equipment and buildings, and prevention and mitigation of core meltdown accidents, etc.

The negative reactivity trips in the Phénix reactor were frustrating in their failure to demonstrate quantitatively the hypotheses put forward for these phenomena, which the analyses of the ‘final’ tests will attempt to do. However, all the scenarios listed when analysing negative reactivity trips and which are likely to lead to reactivity incidents that are potentially harmful to safety should be reviewed in relation to the future design of SFRs. In particular, it will be important to eliminate completely, or failing that, limit the sources of inert gas and fluid that can be vaporized or broken down into gas in contact with the primary cooling system, the risk of rapid rise of one or more control rods, failures in the resistance of core support elements and the consequences of an earthquake (or any other tremor) on core reactivity.

The absence of extremely diverse redundancy to control reactor reactivity, such as the soluble boron in water reactors, means paying special attention to the reliability of the function of introducing anti-reactivity in the core. Constructive provisions should thus minimize the risks of control rods and their mechanisms jamming, especially those due to sodium aerosols and fuel subassembly deformations. The proliferation and diversification of shutdown systems should respond to a probability based approach, where electronic source common mode failures are estimated correctly. Nevertheless, the implementation of a shutdown system meeting the major requirements for reactor shutdown probability (earthquake, passivity, diversification, etc.) should be supported by an analysis of its reliability to prevent it from disrupting power plant operation.

A set of already proven materials is available for future reactors, but improvements are still sought. The materials potentially usable in the future SFRs should be studied, as appropriate, to gain solid knowledge, in particular of their optimum limits and conditions of use (especially with respect to normal and accident operating temperatures), their compatibility with sodium and all the other media they may encounter during maintenance operations, their industrial implementation (forming, welding, etc.), their long term behaviour under load and, where applicable, irradiation, their resistance to corrosion (especially in the presence of aqueous sodium hydroxide) and their cracking mechanisms. R&D work should also focus on consolidating knowledge of corrosion under insulation (or make this corrosion impossible), developing the concept of leak before break and applying them to the materials likely to be used.

A major part of subassembly handling is done ‘blind’ under the opaque sodium, without the direct or indirect line of sight of operators representing a means, albeit fallible, of checking. Alternative means must therefore be designed to ensure that, at any time, the correct subassembly is moved by the correct machine following the correct obstacle free routing as far as the correct location where it is inserted correctly. There are detection resources (limit switches, ultrasounds, thermocouples, etc.), and operating and monitoring PLCs can be produced. The design of future SFRs will aim to ensure their reliability.

Lastly, the safety analysis of how to use non-radioactive intermediate sodium systems must be revised, in agreement with the Nuclear Safety Authority. At the outset of the Phénix and Superphénix projects, this risk had been dealt with in exactly the same way as in the chemical industries. However, firstly, accidents implicating a large quantity of sodium are likely to affect the containment of radioactive products (attack on barriers) and secondly, French regulations change and now require an integrated approach to risks (safety, radiation protection, environment, etc.). It will no doubt be necessary to ‘eliminate practically’ (i.e. by qualified provisions) all the large sodium fires, violent sodium–water reactions and sodium–water–air reactions.

In terms of costs and investment protection, efforts should be made to make general savings in the design, construction and operation of future SFRs with respect to previous projects. To sum up, being ‘simple and robust’ is the key, which is not always easy to put into practice.

The maintenance and inspection operations in Phénix and Superphénix have been onerous. It could be said that these costs are acceptable for an experimental reactor, the first in the world. When, on the other hand, attention is focused on developing a type of electricity generating reactor, competitive with other production methods, nuclear or otherwise, a much better performance must be targeted, especially in terms of ease of implementation and implementation timeframes, as was underlined by the Nuclear Safety Authority after the first incidents at the Creys-Malville power plant. For this reason, emphasis in the first stages of developing future SFRs was placed on the options for inspecting and repairing the reactor structures and on performances, timescales and costs relating to the various reactor designs and the different control methods which could be envisaged.

In general, it should theoretically be possible to control and inspect any component or structure where failure might harm the operation or safety of the power plant. Checks involving specific sodium related difficulties should be harmonized with regulatory practices in particular (and vice versa if possible). The in-service monitoring, inspection (external or intrusive methods, during manufacture or in service) and repair methods should be determined during the first stages of the facility design and should influence it repeatedly, if appropriate.

It should be possible to replace or repair any equipment which may break down, fail (cracking, etc.) or simply for which doubt exists during reactor operation. It is important to stipulate that providing for inspection, replacement or repair of a component in the design phase means that all measures are taken for it to be achieved (cost, timescale and quality). It is only at project completion that the investor/plant operator can decide whether to take the strategic risk of certain 'blind alleys', provided there are adequate guarantees (extra controls at fabrication, additional safety margins, etc.).

The requirement to control, inspect and replace or repair certain internal reactor structures may lead to planning complete unloading of the core during operation. It is, therefore, important to make sure that this unloading is possible, sufficiently short (to limit production losses) and reversible (to avoid 'losing' a partially irradiated core).

Materially, the operations (in-service monitoring, maintenance, replacement, repair, etc.) require space around components. This recommendation should be granted with the requirement to reduce construction costs and, therefore, unnecessary areas. More generally, the constraint associated with the maintenance of the Phénix and Creys-Malville power plants had not been sufficiently considered in the design. Significant progress should be made in this field to reduce operating costs in future reactors.

Lastly, the Phénix and Superphénix reactors were designed, particularly from the viewpoint of safety margins, for total operating times of twenty and thirty years, respectively. For the future electricity generating SFRs, the service life requirement will be at least sixty years, as with the EPR, to ensure an attractive return on investment. This will, in particular, feature in the choice of materials, relevance of modelling (ageing), optimization of various reactor characteristics (temperatures, etc.), maintenance options (half-life replacement of some components?) as well as in the qualification programmes for materials and equipment.

5. CHALLENGE

The fact that this paper, particularly the last part, focuses on potential progress must not overshadow the extremely valuable knowledge that the operation of Phénix and, to a lesser extent, Creys-Malville have brought to the areas of fuel and cores, sodium technology and various compatible steels and components, and to the operation of an SFR.

Obviously, progress can still be made, firstly to raise safety to the same level as third generation reactors and secondly, to improve the performance of the fuel, reactor and electricity generation. The challenge in years to come is to design the future SFRs to meet equivalent requirements to those for water reactors (availability, safety, cost and financial risk) with sustainable development as well (savings in natural nuclear materials, nuclear waste management and, in particular, minor actinides).

Analysing the operating feedback can be used, among other things, to determine the improvements to be made to the SFR design as implemented in Phénix and at Creys-Malville, then in the EFR project, as well as the innovations sought. This is the purpose of the current R&D carried out by CEA, AREVA and EDF.

This paper ends with an issue from the current period, which is fundamental: passing on the knowledge and skills acquired in the past to the men and women who will develop, design, construct and operate the future SFRs. We have a cognitive database and our current responsibility is to make it bear fruit.

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THE LAST TWENTY YEARS OF EXPERIENCE WITH FAST BREEDER REACTORS: LESSONS LEARNT AND PERSPECTIVES

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Abstract

India has made significant achievements in the design and development of sodium cooled fast breeder reactors over the last twenty years. Attaining a maximum burnup of 165 GW·d/t for the plutonium-rich carbide fuel without any cladding failure, coupled with excellent performance of sodium components, including primary pumps, heat exchangers and steam generators over the last 24 years, reprocessing of carbide fuel with a burnup of 150 GW·d/t and engineering tests performed for validating the plant dynamics computer codes, are the achievements from the successful operation of the 40 MW(th) capacity loop type fast breeder test reactor. Indigenous design of the 500 MW(e) Prototype Fast Breeder Reactor (PFBR), executing high quality multidisciplinary R&D and successful manufacturing and erection of large dimensioned thin walled shell structures are the achievements in PFBR development. These achievements, apart from providing confidence in the PFBR project, are instrumental for the design of innovative future reactors. National and international collaboration established with R&D establishments and academic institutions would go a long way towards helping India to attain world leadership by 2020.

1. INTRODUCTION

India is aiming to reach at least a per capita energy consumption of about 2400 kW·h/a (current world average) with an 8% growth rate by 2031–2032. This calls for an electricity generation capacity of 778 GW(e) by 2031–2032 (fourfold increase in twenty years) and tenfold growth over the next fifty years. The nuclear share would be about 25%. Although India has limited uranium resources, it does have abundant thorium resources. The uranium resources reasonably assured plus inferred in India amount to 94 200 t or <2% of the world's resources. However, the thorium resource in the country is 225 000 t (the second largest reserves in the world), which has an energy potential of 155 000 GW(e)/a. The uranium resource available in the country can feed a 10 GW(e) capacity pressurized heavy water reactor for ~50 years with a thermal efficiency of 30% (first stage). The available uranium can also supply 275 GW(e) for about 200 years when used in a fast breeder reactor (FBR) after reprocessing (second stage). Thorium can feed 275 GW(e) capacity power plants for about 550 years. Further, FBRs are essential for converting thorium to ^{233}U, which is required for the third stage. FBRs would provide critical liquid metal technology and high temperature design inputs for the future ADS, fusion and high temperature reactor systems. These apart, they can provide electricity at competitive costs over long periods. Hence, FBRs are essential for the realization of a targeted nuclear share of about 25% (total 1250 GW by 2050) with the limited uranium available.

The fast reactor programme was started in India by the construction of the Fast Breeder Test Reactor (FBTR) at Kalpakkam. The FBTR is a sodium cooled loop type 40 MW(th)/13.2 MW(e) experimental reactor which was commissioned in 1985 and which uses a unique plutonium-rich carbide fuel. The experience gained in the construction, commissioning and operation of the FBTR as well as the 400 reactor-years of worldwide FBR operational experience, extensive experience with MOX fuel, 30 years of focused R&D involving extensive testing and validation, material and manufacturing technology development and demonstration, and peer reviews and synergism among the Department of Atomic Energy (DAE) and R&D institutions and industries have provided the necessary confidence to launch a prototype FBR (PFBR) of 500 MW(e) capacity. The reactor construction was started in 2003 and the reactor is scheduled to be commissioned by 2011. As a follow-up to the PFBR, it is planned to construct three twin units comprising 2 × 500 MW(e) reactors with improved economy and safety during 2010–2020. One twin unit would be constructed at Kalpakkam. It has been realized that for enhanced growth of fast reactors in the country, it is imperative to develop metallic fuelled FBRs, which promise a much higher breeding rate and, hence, beyond 2026, a series of metal fuelled reactors of 1000 MW(e) will be constructed. In this paper, experience gained with FBRs over the last twenty years, R&D achievements and future directions are highlighted [1–4].

2. EXPERIENCE WITH THE FBTR

The FBTR design is the same as that of the Rapsodie-Fortissimo, except for incorporation of a steam generator (SG) and turbogenerator (TG) constructed under the agreement signed with the Commissariat à l'énergie atomique (CEA) in 1969. It has two primary and two secondary sodium loops. Each secondary loop has two once-through serpentine type SGs. A TG and a 100% steam dump condenser to facilitate reactor operation without a TG are also provided. The first criticality was achieved with a small core of 22 fuel subassemblies of MK-I composition (70%PuC–30%UC), with a design power of 10.6 MW(th) and a peak linear heat rating of 250 W/cm. Progressively, the core was expanded by adding SA at peripheral locations. In increasing the core size, and hence the reactor power, carbide fuel of MK-II composition (55%PuC–45%UC) was inducted in the peripheral locations in 1996. The TG was synchronized to the grid for the first time in July 1997. In 2002, the reactor was operated up to a power level of 17.4 MW(th) by raising the linear heat rating of MK-I fuel to 400 W/cm. So far, 15 irradiation campaigns have been completed. The current core has 50 fuel subassemblies: 27 MK-I, 13 MK-II, 8 MOX, one PFBR test fuel subassembly and one special irradiation subassembly. The performance of sodium systems has been excellent. Sodium pumps have accumulated 600 000 h of cumulative, continuous operation. The SGs have performed without a single leak incident. PFBR test fuel is under irradiation and has experienced 92 GW·d/t burnup and 44% MOX fuel has been inducted in the core. On the basis of the confidence derived from post-irradiation examination (PIE) results, a peak burnup of 165 GW·d/t has been achieved in a phased manner. The performance of reactor systems, sodium systems, control rod drive mechanisms and other safety related and auxiliary systems has been satisfactory. The purity of the primary and secondary sodium has been maintained below the plugging temperature of 378 K. The four sodium pumps and their drive systems have been operating very well.

For the reprocessing of fuel, a pilot facility known as CORAL (compact reprocessing facility for advanced fuels in lead cells) has been commissioned at Kalpakkam. In parallel, a demonstration plant is being set up at Kalpakkam. Fuel with a burnup of 150 GW·d/t has been successfully reprocessed for the first time in the world.

2.1. Major incidents and feedback for the PFBR

In May 1987, as part of reactor physics experiments to find out the reactivity worth of a fuel subassembly, it was required to transfer the fuel subassembly from the third ring to the storage location. During this movement, the foot of the assembly was bent as it was projecting below the guide tube in the transfer position. The gripper assembly, the fuel subassembly, the reflector subassemblies and the guide tube were damaged during the incident (Fig. 1). It took about two years to normalize the system. In this process, a novel cutting tool was developed to extract the guide tube just above the equalizing holes. Appropriate remedial measures, including a mechanical stopper for the fuel handling gripper and redundant interlocks for authorizing plug rotation, were implemented. Proper maintenance and operating procedures for the fuel handling mechanism were evolved. As feedback for the PFBR, an under sodium scanner

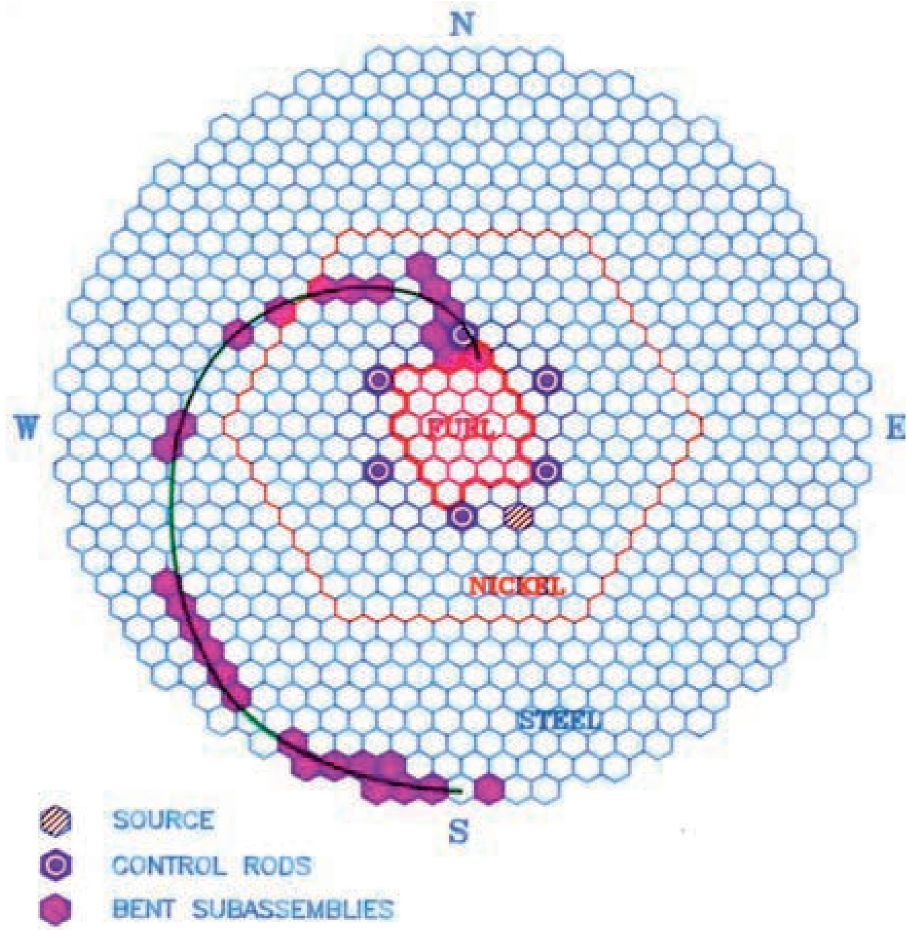

FIG. 1. Damaged subassemblies.

has been developed to ensure that there is no physical projection at the core top before operation of rotatable plugs, apart from incorporating interlocks.

A primary sodium leak (~75 kg) occurred in the purification circuit in April 2002 resulting from a leak from a ligament of the valve body. The defective valves were replaced and the system was brought back into operation within three months. The biological shield concrete surrounding the reactor is cooled by demineralized water flowing through the carbon steel coils. Water leaks were observed resulting from crevice corrosion in the socket welds and these were chemically sealed on-line. To detect such leaks at an early stage, accurate level monitoring was provided. Draining provision was provided to drain out water and

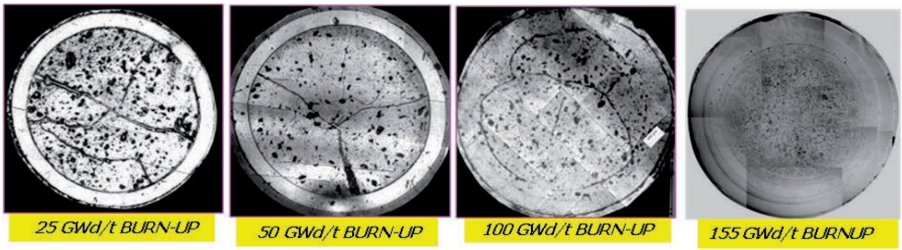

FIG. 2. Ceramographs of fuel–cladding cross-sections at various stages of burnup.

prevent its entry to the A1 cell. In the PFBR, all circumferential welds between pipe to pipe configurations embedded in concrete are butt welds of radiographic quality. Kerb walls are provided around the reactor vault for leakage collection and dewatering provision exists. The expansion tank 'L/D' ratio is chosen to provide good sensitivity for early detection of leakage.

2.2. PIE

Since the high plutonium content carbide fuel was expected to have a high fuel swelling rate, inputs from PIEs were crucial to increase the burnup and linear heat rating beyond the initial design values. PIE was carried out at different stages of burnup, starting with experimental fuel pins, in order to understand the beginning of life performance and after burnups of 25, 50, 100 and 155 GW·d/t. No fuel pin failure has occurred so far, indicating the excellent performance of carbide fuel. The fuel has operated at a peak linear heat rating of 400 W/cm. Figure 2 shows the comparison of photomosaics of fuel–cladding cross-sections at the centre of the fuel column at different burnups. The thermomechanical analyses based on the PIE results indicate that the burnup could be extended marginally to 170 GW·d/t.

3. DESIGN AND DEVELOPMENT OF THE PFBR PLANT AND THE RELATED FUEL CYCLE FACILITY

The PFBR is a pool type reactor with two primary and two secondary loops with four SGs per loop. The nuclear heat generated in the core is removed by circulating sodium from the cold pool at 670 K to the hot pool at 820 K. The sodium from the hot pool, after transporting its heat to four intermediate heat exchangers (IHXs), mixes with the cold pool. The circulation of sodium is maintained by two primary sodium pumps, and the flow of sodium through the

IHXs is driven by a level difference (1.5 m of sodium) between the hot and cold pools. The heat from the IHXs is, in turn, transported to eight SGs by sodium flowing in the secondary circuit. Steam produced in the SG is supplied to the TG. In the reactor assembly, the main vessel is the important component which houses the entire primary sodium circuit, including the core. The sodium is filled in the main vessel with certain free surfaces, blanketed by an argon filled space. The inner vessel separates the hot and cold sodium pools. The reactor core consists of about 1758 subassemblies, including 181 fuel subassemblies. The control plug, positioned just above the core, houses mainly 12 absorber rod drive mechanisms. The top shield covers the main vessel and supports the primary sodium pumps, IHXs, control plug and fuel handling systems. The PFBR uses MOX. For the core components, 20% cold worked D9 material (15%Cr–15%Ni with Ti and Mo) is used to provide better irradiation resistance. Austenitic stainless steel type 316 LN is the main structural material for the out-of-core components and modified 9Cr–1Mo (grade 91) is chosen for the SGs. The PFBR is designed for a plant life of 40 years with a load factor of 75%, which would be increased gradually up to 85%.

A closed fuel cycle with complete recovery of fuel material from the irradiated fuel and the recycling of the fuel are important components for the sustainability of the fast reactor programme. By adopting co-extraction of uranium and plutonium, with optimized decontamination factors for fission products, closed fuel cycles with realizable strategies and technologies could be established and contribute towards achieving economy, safety and societal acceptance. The co-location of the fuel cycle facility (fabrication, reprocessing and waste management) along with reactors would minimize the cost of the energy, allow better physical control of the fissile material and reduce transport risks. This philosophy will, therefore, be adopted in the planning of FBRs at various sites. Simultaneously with the construction of the reactor, the fuel cycle of the reactor has been addressed in a comprehensive manner, and construction of a co-located fuel cycle facility has been initiated, including a dedicated fast reactor fuel cycle facility for reprocessing the high value fissile material remaining in the spent fuel from the PFBR in the form of fabricated fuel pins.

3.1. Challenges and achievements in science and engineering

Regarding the development of the PFBR, significant and high quality scientific input was generated and in this process many sophisticated facilities were developed in the domains of materials, chemistry, thermohydraulics, structural mechanics and component development. Sodium sensors have been developed for measuring ultra-trace levels of dissolved hydrogen, carbon and

FIG. 3. Sodium sensors developed for the PFBR.

oxygen in liquid sodium, in addition to monitoring the level, flow velocity, etc. Figure 3 includes photographs of sodium sensors developed for the PFBR.

Even though the sodium coolant has many advantages, it introduces many challenging thermohydraulics and structural mechanics issues, including high thermal stress and thermal shock on the adjoining metal wall, as well as temperature fluctuations in the metal due to sodium free level fluctuations. Temperature fluctuations are also created in the metal wall due to a special type of phenomenon termed thermal striping, which is caused by a lack of perfect mixing of sodium streams at different temperatures, particularly in the sodium piping. The stainless steel parts when subjected to temperature fluctuations suffer high cycle fatigue damage. Seismic behaviour of interconnected buildings resting on the common base raft, as well as seismic responses of thin-walled vessels, pumps and absorber rod mechanisms, calls for complex numerical and experimental simulation techniques. Addressing these failure modes comprehensively and designing components for long, reliable operation at ~550°C for a design life of 40 years are the most challenging tasks.

High temperature design for long, reliable operation of components operating at temperatures around 820 K for a design life of 40 years, design of mechanisms and rotating equipment operating in sodium and argon cover gas

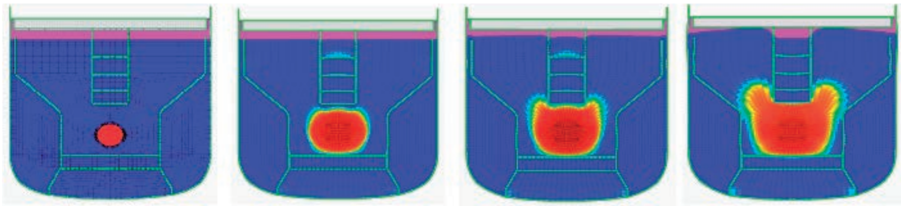

FIG. 4. Mechanical consequences of a CDA.

space, handling the sodium leaks and sodium–water reactions in the SGs, seismic analysis of interconnected buildings resting on the common base raft, seismic design of thin-walled vessels, pumps and absorber rod mechanisms and in-service inspection (ISI) of reactor internals within sodium are a few of the challenging issues addressed in the design. These issues have been successfully resolved through extensive numerical simulations with strong experimental investigations. Peer reviews by national and international expert teams, science based R&D output derived from in-house efforts as well as collaborative projects established through synergism among the DAE, academic institutions, R&D establishments and industries, also add to the high level of confidence in the design. The problems related to sodium have been solved successfully, which has been very well demonstrated by the long and reliable operation of sodium systems in the FBTR, as well as many test loops at the Indira Gandhi Centre for Atomic Research (IGCAR). The science behind sodium fires, sodium–concrete interactions, sodium aerosol behaviour and its effects is well understood and based on extensive numerical and experimental simulation using dedicated test facilities.

The core disruptive accident (CDA) is a very low probability event ($<10^{-6}$/reactor-year) and is considered as a beyond design basis event in the PFBR. The accident scenarios have been understood. On the basis of reactor physics analysis, a mechanical energy release of 100 MJ has been selected, for which the structural integrity of the main vessel, heat exchangers and top shield cover has been demonstrated. Structural analysis for determining deformation and strain in the vessel is carried out using an in-house code (FUSTIN). Figure 4 depicts the sequence of the core bubble expansion and consequent deformation of the main vessel under 100 MJ of mechanical energy release. Subsequently, the sodium release to the reactor containment building (RCB) and finally the temperature and pressure rises in the RCB were estimated and design loadings were defined for the RCB.

The FUSTIN code has been extensively validated using international benchmark problems, i.e. MANON and MARA (France), COVA and CONT (United Kingdom), and TRIG (India) series. TRIG tests were conducted to

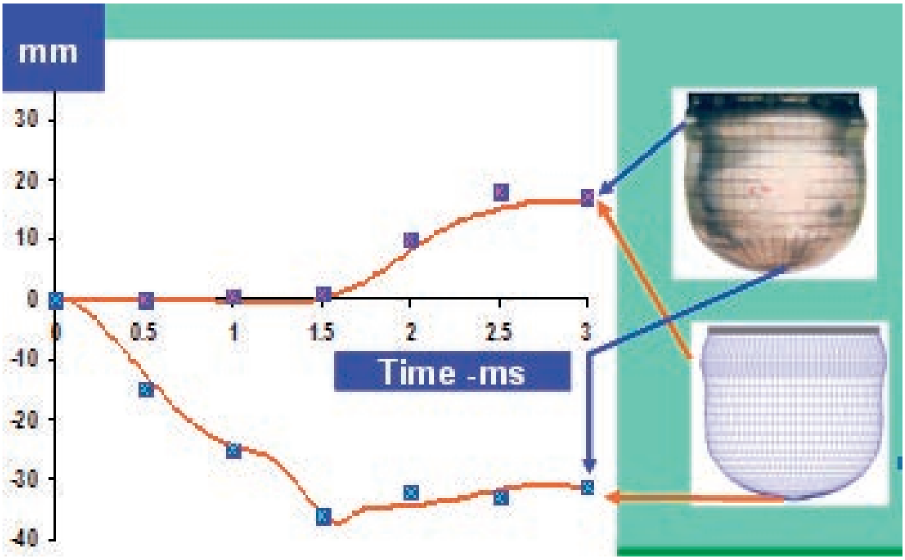

FIG. 5. Transient response of main vessel model under simulated CDA loading.

generate data for the validation of the FUSTIN code. Figure 5 shows one typical validation result. Further, based on a 1:13 scale model testing and evaluation, the structural integrity of the IHX and decay heat exchangers was demonstrated and the sodium leakage through top shield penetrations was estimated. With such extensive numerical and experimental investigations, the structural integrity of the primary containment, as well as that of the RCB, is ensured with a high level of confidence under various CDA loadings.

ISI is crucial to ensure the long, reliable operation of the fast reactor over a lifespan of 60 years. The design stage incorporates all such aspects which make repair feasible, as warranted by the ISI campaigns. The most important aspect of ISI and repair is to carry out the activies in radioactive sodium at a temperature of about 200°C. ISI systems span a range of activities, namely: development of sensors and manipulators that can operate in harsh environments and high quality image processing systems, remote controlled robots to inspect the main vessel completely by travelling over the small intervessel space between the main vessel and the safety vessel and, finally, demonstration of the reliable operation of these devices. Figure 6 shows the prototype vehicle developed for the main vessel inspection, along with a 3-D virtual model and simulation to implement in a 3-D environment, which allow the planning and visualization of the motion of the ISI device in the annular interspace to be accomplished. A remote field eddy current testing technique for ISI of the SG tubes has been developed at IGCAR and this will significantly improve the reliability of SGs.

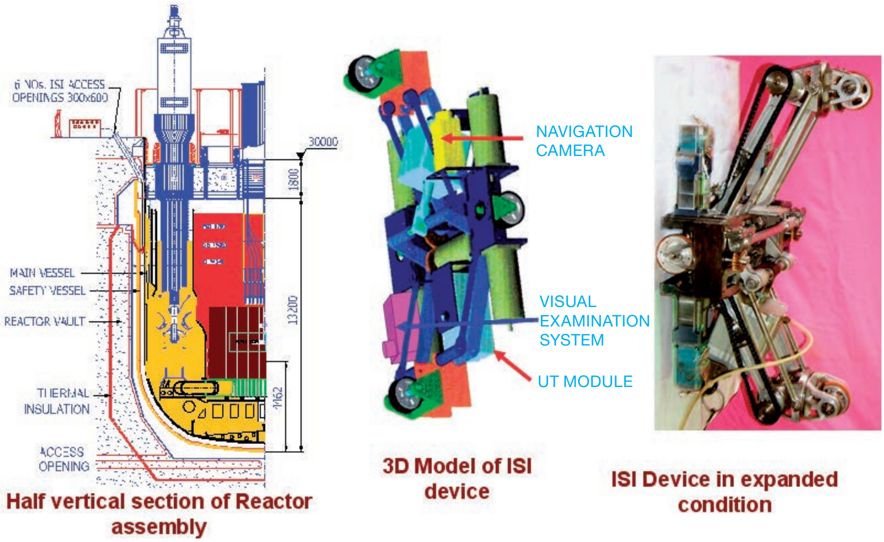

FIG. 6. ISI vehicle for main and safety vessels.

3.2. Challenges and achievements in technology

To eliminate the concern of induced radioactivity, nickel based alloys have been chosen for the hardfacing of PFBR components. The nickel based alloy is highly susceptible to cracking and it was required to deposit this alloy on components of very large dimensions without any cracks. On the basis of extensive mock-up studies and simulations, this challenging task has been completed for the bottom plate of the grid plate assembly within a few hours without generating a single crack. Hardfacing of the inner surface of the grid plate sleeve, in which the fuel subassembly rests, was another critical task that required development of indigenous technology. Figure 7 shows the hardfacing process carried out on the grid plate.

Apart from hardfacing technology, development of large-sized bearings, inflatable seals, high temperature fission chambers, manufacture of large-sized thin-walled vessels made of stainless steel to tight form tolerances and machining and assembly of the grid plate and SGs to close tolerances are some of the challenging issues that have been successfully resolved through detailed technology development exercises. In particular, for the large diameter thin vessels, the major manufacturing challenges are posed by: (i) basic plates, which should not have any defects such as laminations (high quality control is essential); (ii) large lengths of welds while integrating individual petals; (iii) stringent control of manufacturing deviations, such as form tolerances

FIG. 7. Hardfacing process on grid plate.

(<½ thickness), verticality and horizontality (<±2 mm); and (iv) high quality welds and low residual stress, which should be achieved without any heat treatment. In order to 'sensitize' Indian industries and assess manufacturing tolerances that can be achieved by them, elaborate manufacturing technology development works were undertaken prior to the start of construction. Figure 8 shows a few components which have been manufactured through the technology development exercises. The level of confidence of long delivery components such as the main vessel, inner vessel, absorber rod drive mechanisms and SGs on the quality and time schedule has been raised through the manufacturing technology development exercises.

3.3. Achievements in construction

The concept of an interconnected building has been adopted for the 'nuclear island' of the PFBR. The nuclear island extends over a 100 m × 92 m area with very tall buildings; the highest among these is the RCB, which is about 72 m tall. With a base raft thickness of 3.5 m, the civil construction of the nuclear island interconnected buildings involves the pouring of 35 000 m^3 of concrete.

FIG. 8. Components manufactured under technology development.

Completing construction of the PFBR before 2010 requires many challenging and innovative ideas to be implemented, both in construction and in management policies. It is required to carry out civil construction and equipment erection in parallel, which involves using state of the art erection equipment and construction methodologies and highly optimized construction sequences. Erection of very large dimensioned and slender FBR components with very stringent dimensional accuracies (a typical tolerance to be achieved on horizontality over 15 m is less than ±1 mm) is the most challenging task to be completed for the first time in the country. Transport of thin shell structures from the site assembly to the support locations is another challenging activity in the construction. With systematically planned mock-up trials, there is high confidence that the components would be erected successfully, meeting all the specified erection tolerances. This has already been demonstrated through the successful erection of the safety vessel in June 2008 and the main vessel in December 2009 (Fig. 9). Innovative handling structures were designed and tested for transporting the slender structures without causing any permanent deformation.

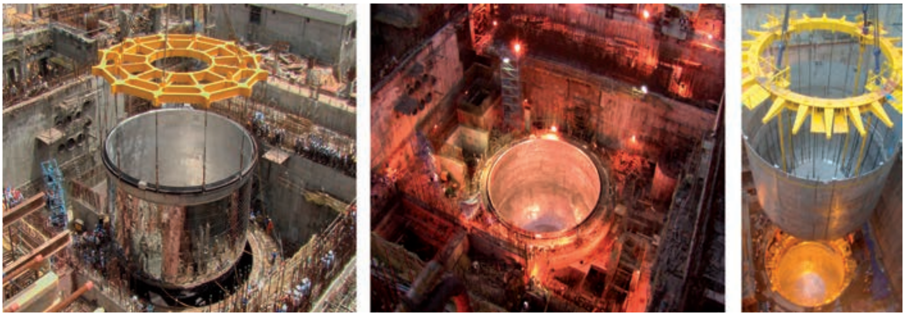

FIG. 9. Successful erection of safety and main vessels.

4. APPROACH FOR THE DEVELOPMENT OF FUTURE FBRs

Enhanced safety and improved economics are twin objectives. The means to achieve economy has been quantified: increased design life to 60 years, design load factor of 85%, construction time 5 years, reduction in special steel specific weight requirement by ~20% and enhanced burnup in a phased manner (target: a unit energy cost comparable to that of fossil fuel power plants). The mechanisms to achieve enhanced safety are being assessed: elaboration of ISI and repair provisions; increased reliability of shutdown systems, decay heat removal system and in-vessel purification system; and the innovative post-accident heat removal provisions. Science based technologies and breakthroughs at interfaces in science and science with engineering are to be harnessed for sustainable technological solutions. Extensive involvement of industries from the developmental stages and giving due consideration to the innovative features conceived in the fast reactors under international projects (Generation IV and INPRO) are the strategies adopted for future reactor development.

5. COLLABORATION AND HUMAN RESOURCE DEVELOPMENT

In order to meet the urgent and growing need of scientific human resources for the correct training, a training school for engineering and science post-graduates at IGCAR was established. By virtue of the multidisciplinary expertise, IGCAR has established itself as a research centre of national and international repute, not only in the primary areas of fast reactor technology but also in many associated areas. For achieving the mission objectives of the centre, IGCAR has entered into collaboration with a number of educational and research institutions. IGCAR has been an active participant in the International Working Group on Fast

Reactors and other IAEA activities of interest to FBR programmes. IGCAR has also been actively participating in international cooperative IAEA research programmes in the areas of reactor engineering, reprocessing and safety. The early collaboration of IGCAR with the CEA dates back to 1969, when the FBTR was conceived to be built by adaptation from the French fast reactor Rapsodie with several design modifications. IGCAR re-established collaboration with the CEA in 1989 to exchange computer codes in the field of thermohydraulics and structural mechanics. Under this collaboration, IGCAR received the CASTEM 2000, PLEXUS and TEDEL codes for structural mechanics analysis from the CEA. Recently, the DAE has established collaboration with the CEA over a wider spectrum of subjects of interest, in particular FBR safety, which is of direct relevance to IGCAR.

6. SUMMARY

India enhanced the R&D activities during the last twenty years for the rapid realization of its nuclear potential through the FBR. Attaining a maximum burnup of 165 GW·d/t for the plutonium-rich carbide fuel without any cladding failure and with excellent performance of sodium components, including primary pumps, heat exchangers and SGs, for the last 24 years and reprocessing of carbide fuel with a burnup of 150 GW·d/t are the achievements from the successful operation of the 40 MW(th) capacity loop type FBTR. Indigenous design of the 500 MW(e) PFBR, accomplishing advanced R&D activities and successfully manufacturing and erecting large dimensioned thin-walled shell structures are the achievements in PFBR development. These achievements, apart from providing confidence in the PFBR project, are instrumental for the design of innovative future FBRs. National and international collaboration will go a long way towards helping India attain world leadership by 2020.

ACKNOWLEDGEMENTS

The contributions of colleagues and collaborating individuals and organizations from India and abroad are sincerely acknowledged.

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THE LAST TWENTY YEARS OF EXPERIENCE WITH FAST REACTORS IN JAPAN

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Abstract

Fast reactor development experience gained in Japan in the last twenty years is summarized in this paper. In this twenty years, the safety, reliability and economic goals of fast reactors have become more ambitious than in the past. However, twenty years of progress have shown that the domestic commercialized sodium cooled fast reactor (SFR) concept, the Japanese SFR, could achieve those targets discussed in the Feasibility Study on Commercialized Fast Reactor Cycle Systems (FS) and the Fast Reactor Cycle Technology Development (FaCT) projects. The Monju prototype fast breeder reactor is finally going to restart by the end of this Japanese fiscal year (March 2010) and will take on the role of a technology and human resource development centre from both a domestic and an international point of view.

1. INTRODUCTION

Japan, which is poor in natural resources, has been developing fast breeder reactors (FBRs) with an unshakable national policy in order to secure domestic long term energy resources. Nuclear power without gross carbon dioxide emission is thought to be one of the strong solutions to global warming. To provide sustainable nuclear power, uranium resources and handling of high level radioactive wastes are major issues. FBR development has been revived internationally, since FBRs have the potential to solve the two issues by breeding fuel and burning transuranic elements, which are toxic and have long lives, as with high level wastes. The Framework for Nuclear Energy Policy formulated by the Atomic Energy Commission of Japan in October 2005 states that striving for the commercial use of FBRs from around 2050 is one of the guidelines for the promotion of nuclear power generation in the future. As regards FBR development, this has been stepped up from an experimental reactor (Joyo) to a prototype reactor (Monju) and currently R&D is being promoted on a demonstration reactor leading towards development of commercial reactors.

A council, which was established in 2006 by administrative authorities, utilities, vendors and the Japan Atomic Energy Agency (JAEA), aiming for a smooth transition to a demonstration process, agreed that it was necessary to carry out FBR R&D efficiently under a structure with clear responsibility. Then, Mitsubishi Heavy Industries was selected as a core company in April 2007 for R&D up to the startup of the basic design for the FBR demonstration reactor. In order to concentrate on a responsibility, an authority and an engineering function on FBR development, Mitsubishi Heavy Industries founded a new company, Mitsubishi FBR Systems. The framework for promoting FBR development in Japan was, therefore, established, based on coordination between the JAEA and Mitsubishi FBR Systems, where cooperation with vendors and universities is a key element.

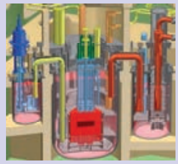
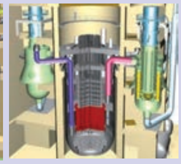
2. EXPERIENCE FROM FAST REACTOR (FR) DEVELOPMENT IN JAPAN

2.1. Joyo experimental FR

The Power Reactor and Nuclear Fuel Development Corporation (PNC, currently JAEA) started construction of the experimental Joyo FBR in 1970. The purposes of the construction were, firstly, to design, construct and operate a mixed oxide (MOX) fuelled loop type sodium cooled FBR (SFR) with domestic technologies and then to accumulate technical knowledge, and secondly, to carry out necessary irradiation tests of fuels and materials for FBR development. The history of the Joyo operation was reported in the previous FR conference as having accumulated 14 years of operating experience by 1991 [1]. In this paper, the 30 years of experience gained with Joyo is summarized [2].

The major specifications of Joyo are shown in Table 1. Joyo achieved first criticality in April 1977 as a Mk-I breeding core. The thermal output was 50 MW which was then increased to 75 MW. Joyo is a loop type SFR with two main cooling systems and one auxiliary cooling system. Since neither a steam generator (SG) nor an electric generator is installed in Joyo, reactor heat is released to the air by the dump heat exchangers in the main secondary cooling systems. The breeding performance of the Joyo Mk-I core was confirmed with a breeding ratio of 1.03. In 1981, the Mk-I breeding core was modified into the Mk-II core as an irradiation test bed and this achieved criticality in 1982. Then, many irradiation tests were carried out for both Monju and a larger reactor. The Joyo core was upgraded again from 2000 to 2003 to improve irradiation performance as the Mk-III core. The core thermal output was increased from 100 to 140 MW. Intermediate heat exchangers (IHX), dump heat exchangers and secondary pump motors were entirely replaced according to the thermal output increment [3]. The Mk-III core performance tests were successfully conducted after the modification [4]. Major Joyo achievements are listed as follows:

TABLE 1. MAJOR PARAMETER CHANGES

Item	Joyo	Monju	DFBR	JSFR
Reactor				
Electric output		280 MW	660 MW	1500 MW
Thermal output	140 MW (Mk-III)	714 MW	1600 MW	3570 MW
Fuel	MOX	MOX	MOX	MOX
Configuration	Loop	Loop	Loop	Loop
Number of loops	2	3	3	2
Output per loop	70 MW	238 MW	533 MW	1785 MW
Sodium temperature	500°C	529°C	550°C	550°C
Piping material	SS304	SS304	SS316	Mod. 9Cr-1Mo
Operation cycle	60 (Mk-III)	6 months	12 months	26 months

— Breeding demonstration (Mk-I);
— Plutonium fuel recycle demonstration (Mk-I to Mk-II);
— Natural circulation test (Mk-II);
— Failed fuel identification test (Mk-II);
— Advanced fuel irradiation (nitride, carbide, etc.) (Mk-II);
— High burnup irradiation (achieved 140 GW·d/t) (Mk-II);
— On-line instrumented irradiation device development (Mk-II);
— Power-to-melt test (Mk-II);
— Material irradiation test (over 4000 test pieces) (Mk-I to Mk-III);
— Minor-actinide-bearing (MA-bearing) MOX fuel irradiation (Mk-III);
— Demonstration of self-actuated shutdown system (Mk-III);
— In-pile creep rupture experiment of ODS steel (Mk-III).

Details of the above items are described in Ref. [2]. Joyo was successfully operated without any major trouble, accumulating 70 798 h of operation up to 2007. However, Joyo has been shut down because of the disconnecting failure of the instrumented type irradiation test device MARICO-2, which occurred during the 15th annual inspection which started in May 2007. Details are described in Ref. [5].

2.2. Monju prototype FBR

To demonstrate that the loop type SFR can be used as a power reactor with the experience of Joyo applied, PNC started construction of Monju in October 1985 and completed it in April 1991. Plant characteristics and construction were reported in the previous FR conference [6]. The major specifications of Monju are shown in Table 1. The Monju core is a MOX fuel type, as is the Joyo core. The electric and thermal outputs are 280 and 714 MW, respectively, and the thermal output is five times greater than that of the Joyo Mk-III (40 MW). Monju has SGs and a turbine generator to demonstrate the capability of an SFR to generate electricity. There are three circuits in the main cooling system, and an evaporator and a superheater were installed in each secondary cooling system. For the installation of the SG system, two 50 MW prototype SGs were manufactured and operated from 1974 to 1987 [7]. For the fuel handling system, an ex-vessel storage tank was newly installed, while Joyo provides in-vessel storage positions. The operation cycle is 6 months, which is much longer than that of Joyo's range of 45–70 d. Monju achieved first criticality on April 1994. However, a sodium leak incident occurred in the secondary coolant system on 8 December 1995 and its operation has been shut down since. After comprehensive reviews on the validity of FBR development in Japan, as well as on Monju safety, the plant modification work against sodium leakage was carried out in 2005–2007. Then, Monju was checked for its entire plant soundness and is under preparation, including conducting the container vessel leak rate test, towards the restart, which is planned by the end of the Japanese fiscal year 2009 (March 2010). As of October 2009, to improve the anti-seismic margin, vibration absorption structures are being incorporated into the ventilation chimney and tide gauges to the seawater intake, in parallel with preparations towards the restart.

The original mission of Monju was to demonstrate the reliability of the FBR power plant through its operation and to establish sodium handling technologies. Monju is now positioned as a core function in the energy R&D centralization plan that Fukui prefecture is promoting and will be used as one of the main fields of R&D leading towards FR commercialization.

Recently, data on domestic and foreign problems arising in FRs and LWRs have been collected to provide the risk communication material entitled The Expected Problems and Countermeasures in Monju. About 900 pieces of information were collected, including 413 cases from foreign FRs, 15 from Joyo, 54 from Monju and 390 from LWRs in Japan. The FR problems were categorized according to problem type and equipment. The results showed that problems due to sodium account for about 50% of those arising in FRs (e.g. sodium leakage, sodium sticking, sodium blockage and SG tube failure, as shown in Fig. 1).

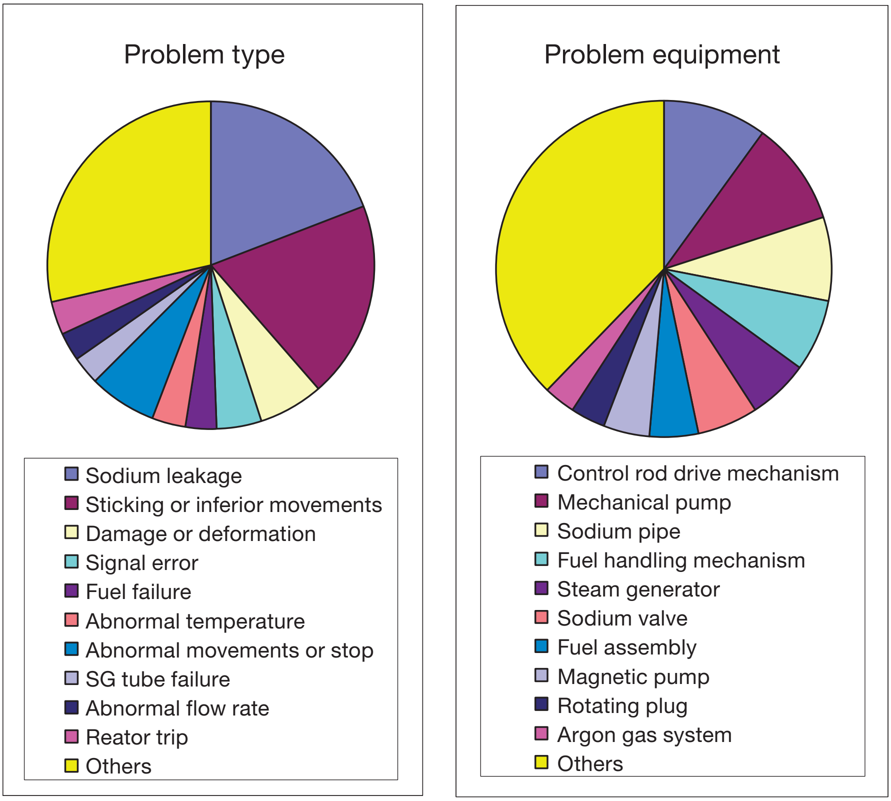


FIG. 1. SFR problem types and equipment.

Some problems arising from specific designs which are different from Monju were excluded from the risk communication material. Similar type problems were put together into one combined group. As a result, all collected problems (about 900) were grouped into 13 categories and 122 problems were selected as ‘expected problems and countermeasures in Monju’ as shown in Table 2. According to the selection, risk communication sheets were made for each problem.

The developed risk communication sheets, which are composed of abstracts and details on effects on the environment, countermeasures, disclosure standards and brief figures, have been utilized to promote the safety culture in Monju and have been made available to the public.

TABLE 2. CATEGORIZED PROBLEMS IN MONJU

Group	Number of items
1. Sodium leakage (primary, secondary, ex-vessel storage tank)	16
2. Sodium sticking and blockage or foreign matter in sodium system	14
3. Problem in the measuring and control system	14
4. Problem with movement of equipment	14
5. Steam leakage, water leakage, etc. (seawater, oil, chemicals)	9
6. Deformation and failure (structure, tank, pipe, heat transfer tube, fuel)	8
7. Radioactive leakage (gas, liquid, solid)	8
8. Problems caused by experiments (modified system function test, entire system function tests, system startup test)	4
9. Flames (electrical equipment, oil, welding)	4
10. Problems with electrical equipment (diesel generator, electric transformer, electric generator)	4
11. SG tube failure	2
12. Fuel failure	2
13. Others (human error, injury, natural disaster, etc.)	23

2.3. Research into commercialization of FRs

2.3.1. Demonstration FR designed by utilities

As for the design of a demonstration FR in the next stage after Monju, the Japan Atomic Power Co. became a centre for FR development and initiated a demonstration FBR (DFBR) project in 1984. A target for the DFBR construction cost with the same safety, operability and maintainability levels of an LWR is less than 1.5 times that of an LWR. At the first stage, both of the various sodium cooled reactor configurations were investigated, including the pool, loop and hybrid. Following this, design studies and evaluations were conducted to select the DFBR reactor configuration [8]:

- — Cost reduction design study (1984–1986);
- — Investigation of the FBR system, including innovative technology (1987–1988);
- — Evaluation study of maintainability and reparability (1988);

— Reactor type evaluation study (1989);
— Basic DFBR specification selection (1990).

In the reactor type evaluation study in 1989, both pool and loop type reactors with 1000 MW electric output were conceptually designed and compared from the viewpoint of safety, structural integrity, operability, maintainability, manufacturing and economic feasibility. No significant difference in safety, structural integrity and operability was found between the two configurations. The loop type had an advantage on manufacture and maintainability, while the pool type had an advantage on construction cost, showing that the material amount of the loop type components are 1.06 times larger than those of the pool type.

As a result of the comparative study, the loop type configuration was selected and the DFBR pre-conceptual design was initiated with a MOX fuel core, a top-entry loop and 660 MW(e) output [9]. The major DFBR plant parameters are shown in Table 1. The DFBR configuration was a loop type and had primary cooling system components such as primary pumps and IHXs accommodated in independent vessels and connected by top-entry piping. The piping material is 316 stainless steel instead of the 304 stainless steel adopted for Monju. The following studies were conducted for DFBR development:

— Pre-conceptual design study (1990–1991);
— Conceptual design study (1992–1993);
— Evaluation study of maintainability and reparability (1988);
— Design optimization phase I (1994–1996);
— Design optimization phase II (1997–1999).

As a result of the conceptual design and the phase I optimization, the DFBR showed that the construction cost could achieve the target of 1.5 times the construction cost of a 1000 MW-class LWR. However, during the phase I study, the situation with respect to FBRs changed and the target construction cost was reduced. Additionally, in 1995, the Monju sodium leak incident occurred, resulting in a long plant outage. As a result of the Monju leak experience, further integrity aspects and public acceptance were taken into account when establishing design requirements. On the basis of these new FR situations, the phase II design optimization was conducted from 1997 to 1999 and the construction cost of the phase II optimized design was estimated at around 1.3 times that of the LWR.

2.3.2. Japanese SFR (JSFR)

In 1999, the JAEA launched the Feasibility Study on Commercialized Fast Reactor Cycle Systems (FS) with domestic partners, such as utilities, vendors and universities. The FS targets were more encouraging than in the previous DFBR project considering the domestic FR situation. The FS targets also met the goals of the Generation IV International Forum (GIF) initiated in 2000 by the United States of America. The major FS targets are listed as follows:

- Ensuring safety:
  —Enhancing the prevention capability against core disruptive accident (CDA) initiators by passive mechanisms;
  —Enhancing the mitigation capability against CDA consequences within the vessel without severe energy release.
- Economic competitiveness:
  —Step 1: Achieving a power generation cost comparable to that of future LWRs (150 GW·d/t burnup, over 90% availability);
  —Step 2: Ensuring cost competitiveness in the global market.
- Efficient utilization of resources:
  —Sustainable usage of nuclear energy (breeding ratio of ~1.2);
  —Burning transuranic elements as fuel (MA-bearing up to 5%).
- Reduction of environmental burden:
  —Reducing the amount of radioactive waste for disposal by burning transuranic elements;
  —Using MA and long lived fission product transmutation leads to shortening the time equivalent to the potential hazard (radiotoxicity) of natural uranium of less than 1000 years.
- Enhancement of nuclear non-proliferation:
  —No pure Pu in any FR cycle system.

The FS scope includes various advanced FRs such as SFR, GFR, heavy metal cooled FRs (LFR and LBFR) and water cooled FRs with various fuels (e.g. oxide, nitride and metal, as shown in Table 3). The FS scope also includes advanced SFR concepts with simplified secondary sodium circuits using advanced IHX, advanced energy conversion systems and multipurpose plants. Pre-conceptual design studies on those various concepts and evaluations were conducted, and four concepts were selected for further evaluation in FS phase II from 2001 [10]. From the viewpoint of the SFR configuration, the loop type concept was selected since it succeeded in dramatically reducing construction costs from conventional loop concepts and achieved the same level of

TABLE 3. CONCEPTS INVESTIGATED IN FS PHASE I

Category	System concepts	Selection for phase II
SFR	Loop type (1500 MW) Pool types (1500 MW) × 3 concepts Loop modular (750 MW) Loop modular (500 MW)	Loop type (1500 MW, MOX fuel)
GFR[a]	CO_2 cooled + steam turbine He cooled + steam turbine He cooled + gas turbine (pin fuel) He cooled + gas turbine (particle fuel)	He cooled + gas turbine (1124 MW, nitride particle fuel)
LFR[b] and LBFR[c]	Loop type Large scale pool type Middle scale pool type (LFR) Middle scale pool type (LBFR)	Middle scale pool type (750 MW, LBFR)
Water cooled reactor	BWR type PWR type (heavy water) Supercritical water cooled	BWR type (1356 MW, MOX fuel)
Small reactors	LBFR (50 MW) SFR (50 MW) SFR (165 MW) SFR (300 MW)	
Multipurpose reactors	Hydrogen production plants (methane steam reforming, thermochemical electric hybrid method)	
Advanced SFR	Advanced IHX concept (simplification of secondary system) × 8 concepts Advanced energy conversion system (CO_2 cycle, magneto hydro dynamic, thermoelectric)	

[a] GFR: gas cooled FR.
[b] LFR: lead cooled FR.
[c] LBFR: lead–bismuth cooled FR.

construction costs as the advanced pool types, while maintaining the advantages of the general loop concepts [11].

In FS phase II, conceptual designs of the four selected concepts were conducted and R&D roadmaps for concepts were investigated. Sketches of the phase II concepts are shown in Fig. 2 and a brief summary of the evaluation results are shown in Table 4. The target construction cost was achieved by all four

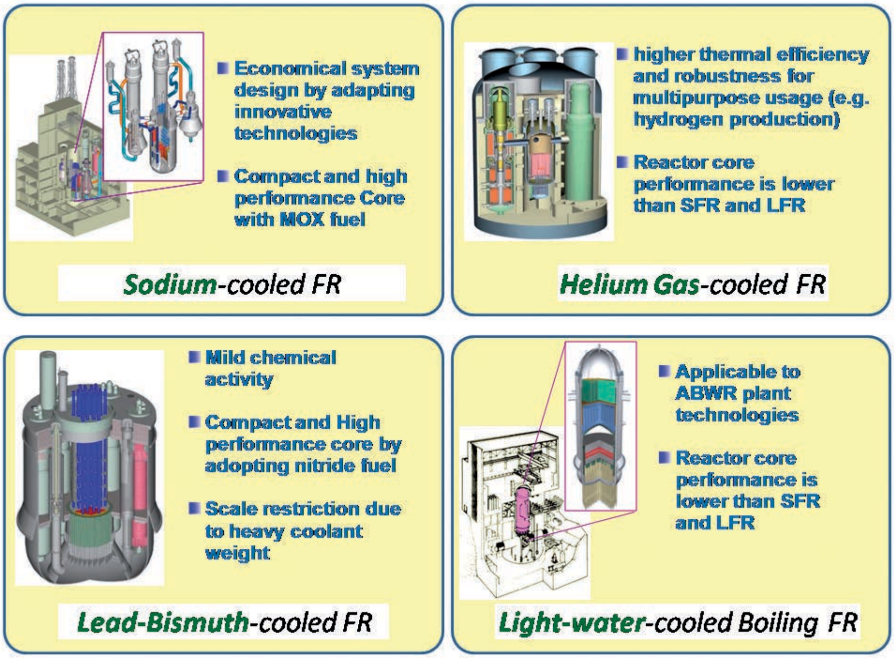


FIG. 2. FS phase II concepts.

concepts and the target core performance was achieved by three concepts, except for the water cooled FR. A rough depiction of the R&D roadmaps of the four concepts is shown in Fig. 3. From the viewpoint of R&D roadmaps, sodium and water cooled FRs held an advantage, since experience and base technology has been accumulated from Joyo, Monju and the advanced boiling water reactor. The lead–bismuth and helium cooled FRs would require an experimental FR and have to overcome several very difficult issues. As a result, the advanced SFR, the JSFR, was selected as the Japanese FR concept and the Fast Reactor Cycle Technology Development (FaCT) project was initiated in 2006 [10, 12].

The FaCT project scope includes construction of a demonstration reactor scheduled to start operating in 2025. Innovative technology selection for the JSFR is currently progressing. Conceptual design of the commercial and demonstration FRs with specific R&D roadmaps should be fixed in 2015. The details of the FaCT project are discussed in Ref. [12]. Figure 4 shows the Japanese FR development steps. The development started from the experimental Joyo FR and the restart of the Monju prototype FBR is anticipated. The DFBR concept was developed with the target equivalent to that of the LWR and now the JSFR is being developed, which could compete with other future energy sources, aiming for commercialization by 2050.

TABLE 4. SUMMARY OF THE FS PHASE II EVALUATION

Coolant	Sodium	Helium	Lead–bismuth	Water
Electric output	1500 MW	1124 MW	750 MW	1356 MW
Coolant temperature	550°C	850°C	445°C	287°C
Safety	SASS[a], CDA energetics free core, in-vessel core catcher	SASS, fuel relocation via bottom of reactor vessel, ex-vessel core catcher	SASS, possibility for recriticality prevention due to fuel dispersion by buoyancy	Possibility for fuel debris retention at the lower head of reactor vessel with neutron absorber
Breeding ratio	1.03–1.10	1.03–1.11	1.04–1.10	maximum 1.05
MA + FP[b] fuel	5%MA+2%FP	5%MA+2%FP	5%MA+2%FP	4%MA
Burnup	150 GW·d/t	123 GW·d/t	155 GW·d/t	88 GW·d/t
Thermal efficiency	42%	47%	38%	35%
Construction cost	90%	100%	100%	100%
Availability	95%	93%	93%	93%

[a] SASS: self-actuated shutdown system.
[b] FP: fission products.

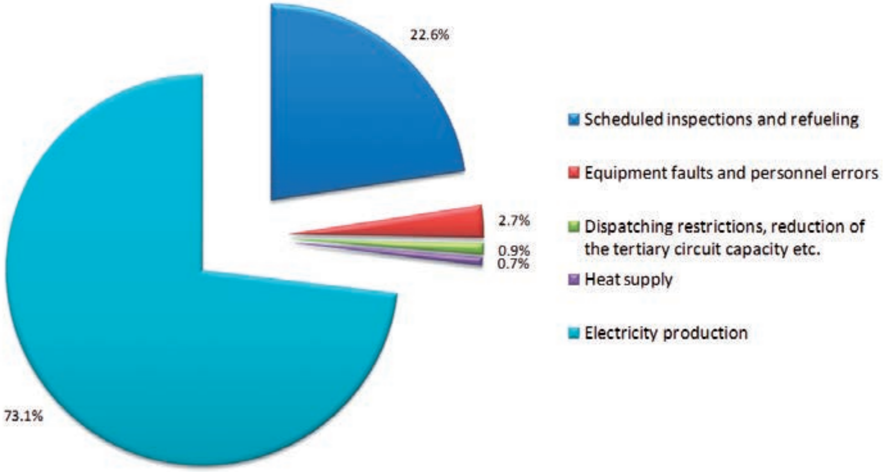


FIG. 3. Roadmaps for FS phase II concepts.

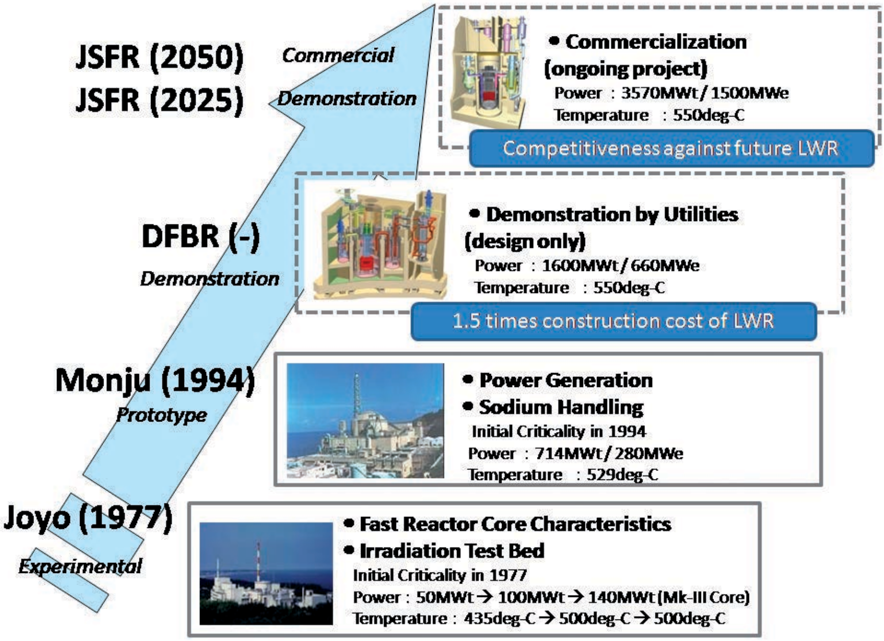


FIG. 4. SFR development steps.

2.3.3. Others

The super-safe, small and simple (4S) FR is one of the SFR concepts suitable for remote area power sources and was proposed by the Central Research Institute of Electric Power Industry and the Toshiba Corporation [13]. The SFRs for remote power sources generally have a long refuelling cycle and passive safety features that take account of the remote circumstances. In the FS, small SFRs, including a 50 MW remote power source [14], were also investigated to reveal their potential. The FS evaluation showed that they could not meet the FS requirement on economic competitiveness as base load power generation. Toshiba and the Central Research Institute of Electric Power Industry continue the 4S development, and the design of a 30 MW thermal output version was completed to commence the US licensing process. Galena in Alaska offered a siting acceptance for 4S in 2004 and the US Nuclear Regulatory Commission initiated a pre-application review for 4S in October 2007 [15].

The University of Tokyo has proposed supercritical pressure water cooled reactors (SCWR), which include thermal (super LWR) and super FR versions [16]. Both super reactors provide compact once-through boiler concepts and high economic performance. Ongoing major development programmes involving the

University of Tokyo, Kyushu University, Tokyo Electric Power Company and the JAEA are part of the super FR project, which includes super FR concept development, thermohydraulic experiments and material development. The SCWR was also nominated as one of the GIF concepts.

3. LESSONS AND TASKS FOR THE FUTURE

3.1. International cooperation

The circumstances of international SFR development have been dramatically changed. From the 1980s to the 1990s, several SFR projects were terminated, such as the Clinch River Breeder Reactor in 1983, the SNR-300 in 1991, the PFR in 1994 and the Superphenix in 1998. Additionally, SFR development was shifted to that of a rather long term project and other coolant FRs were revived. The USA first proposed the Generation IV concept in 1999 and GIF was created in July 2001, currently comprising 12 countries and an international agency (EC). The GIF scope includes six reactor systems: the SFR, GFR, LFR, Molten Salt Reactor, SCWR and the Very High Temperature Reactor. The FR situation changed dramatically in 2006. In January 2006, the French president announced a national project which includes a Generation IV prototype reactor to commence operation in 2020; an SFR is thought to be a strong option for this prototype reactor. In February 2006, the USA proposed the Global Nuclear Energy Partnership, which has grown into an international framework with 25 partner nations committed to pursuing the expansion of clean, sustainable nuclear energy worldwide in a safe and secure manner, while at the same time reducing the risk of nuclear proliferation. In April 2006, Japan announced the FaCT project, which targets constructing a demonstration SFR plant by 2025. The USA and Japan established the Joint Nuclear Energy Action Plan in 2007, including FR development actions. In January 2008, the US Department of Energy, the French Commissariat à l’Énergie Atomique and the JAEA expanded cooperation on SFR prototype development through a Memorandum of Understanding, which established a collaborative framework for the three research agencies to cooperate, with the ultimate goal of deploying SFR prototypes. In order to proceed with FR development efficiently, harmonization of international cooperation is important [17]. Further, to contribute to the global peaceful use of FR development, the JAEA cooperated in a project where Pu derived from dismantled nuclear weapons is burned in a BN-600 in the Russian Federation (the BN-600 vibropac option).

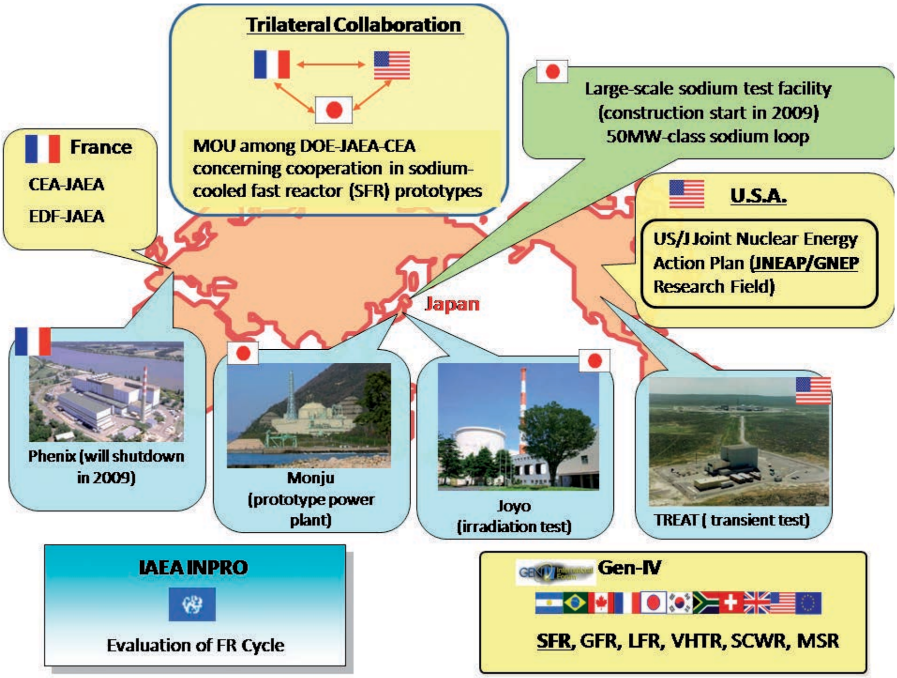


FIG. 5. Overview of international collaboration.

3.2. Development of human resources

Operation of Joyo has provided valuable experience and proved fruitful in the development of human resources. The student training courses using Joyo and related experimental facilities of the JAEA have been initiated to utilize the nuclear facilities and their engineering staffs for educational purposes. The development of student training courses has also been strongly supported by the faculties of nuclear engineering of domestic universities whose curriculum has recently been reduced. The programme covers the reactor physics test analysis of the Joyo core, experiments using the Joyo full scope training simulator, neutron dosimetry, measurement of trace amounts of noble gas and chemical analysis of sodium. The programme has started after a check and review by specialists in university education. It is expected to promote human resource development for the younger generation in the nuclear industry and to strengthen relations between the JAEA and the universities in the area of research.

The International Nuclear Information and Training Center near Monju provides broad educational training courses for domestic and international clients [18]. It has two major facilities: (i) the FR Training Facility, which offers two types of training for sodium handling and maintenance and (ii) the Monju Advanced Reactor Simulator for operation training. A total of 27 training courses comprising: 8 simulator courses, 5 FBR plant system engineering courses, 7 sodium handling courses and 7 maintenance courses are available using these facilities. For international cooperation, an International Reactor Plant Safety Training Course is offered to help spread nuclear safety technology to Asian countries and an International Sodium Technology Training Course is always offered for trainees from China and the USA. The variety of training courses will contribute to the development of engineers who are expected to play a key role in the future development of the FBR.

4. CONCLUSION

The last twenty years of SFR development in Japan have changed dramatically. Since the Monju shutdown in 1995 and the change in the international SFR circumstances during the 1980s and 1990s, the JSFR development situation has been adversely impacted. The safety, reliability and economic targets have become more ambitious. However, the last twenty years of effort have shown that the domestic commercialized SFR concept, the JSFR, could achieve those targets discussed in the FS and in the FaCT project. One important example of progress is the restart of Monju by the end of March 2010. The restart of Monju is necessary from the viewpoints of technology and human resource development. Monju is also expected to be utilized as an international collaboration test bed. With this twenty years of experience, Japan is now ready to develop commercialized FR systems and achieve challenging goals for future energy sources.

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EXPERIENCE GAINED IN THE RUSSIAN FEDERATION ON SODIUM COOLED FAST REACTORS AND PROSPECTS FOR THEIR FURTHER DEVELOPMENT

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Abstract

Experience gained with sodium cooled fast reactors (SFRs) in the Russian Federation over the past 30 years is reviewed. Some statistical data on the operation indicators gained for SFRs worldwide and in the Russian Federation are presented. The basic phases of SFR technology development in the Russian Federation are described. The main research work undertaken on SFRs in the Russian Federation (BR-5/BR-10, BOR-60, BN-600) for justification of SFR technology is highlighted. Priority is given to analysis of operational experience of industrial power unit No. 3 of the BN-600 reactor at the Beloyarsk nuclear power plant and the operation indicators achieved. Statistical information is presented on abnormal events that occurred during operation of the BOR-60 and BN-600 units, and the extent of their influence on the facilities' safety, technical and economic performances. Conclusions on the level of mastery of SFR technology achieved in the Russian Federation are made, based on the review of operation experience, and prospects for their further development are estimated in the light of this experience.

1. INTRODUCTION

Experience gained with sodium fuelled fast reactors (SFRs) in the Russian Federation over the past 30 years and prospects for their further development are reviewed in this paper. This experience includes both results of R&D and design developments on SFRs, and knowledge gained in the construction, commissioning, operation and decommissioning of facilities with SFRs. Beyond a doubt, this experience per se is of extreme value, even the negative aspects. However, eventually, the main objective of acquiring knowledge is to form a base for the development and industrial mastery of safe and reliable SFR technology.

The reactor technologies actually are a summation of diverse technologies. This scope includes: technology for manufacture, operation and reprocessing of fuel; technology for development and operation of structural materials with specified operational characteristics; coolant technology; technology for manufacture and operation of the main SFR equipment; principles and approaches for coordinated control of SFR systems and equipment and ensuring the required safety level, etc. All these interrelated technologies should be developed and mastered to a level that allows justification, construction, operation and decommissioning of nuclear power plants with SFRs in accordance with regulatory specifications and documents currently in force. As a rule, any reactor technology is created on the basis of R&D intended for this purpose and, further, is subject to corrections, with due account taken of the experience gained in the operation of actual facilities. SFR technology also depends on an accumulation of both the results of R&D and the operational experience gained. In turn, the operational indicators of facilities with SFRs are evidence of the quality of the development and the level of industrial assimilation of SFR technology as a whole. Therefore, development and the level of mastery of SFR technology will be the main aspect of the review of experience gained in the Russian Federation on SFRs.

There is no claim for comprehensive analysis of each particular aspect of SFR technology in this paper. Such analysis has already been presented in separate reports related to specific aspects of SFR technology. The objective of this work has been to analyse the level of mastery of SFR technology in the Russian Federation using the examples of operating facilities employing SFR technology, with reference to all domestic power reactors, and to attempt to define the main trends in updating this technology in the future and the prospects for its further development.

2. PHASES OF MASTERING SFR TECHNOLOGY

R&D work on SFRs was initiated in the former USSR in the second half of the 20th century, and has continued for over fifty years. The work has resulted in the following facilities:

(a) Research fast reactor BR-5/BR-10 (IPPE, Obninsk);
(b) Experimental fast reactor BOR-60 (RIAR, Melekess/Dimitrovgrad);
(c) Prototype reactor facility (RF) with BN-350 reactor (Mangyshlack power combine, Shevchenko/Aktau, Kazakhstan);
(d) Industrial power unit with BN-600 reactor at the Beloyarsk nuclear power plant in Zarechny;
(e) Design of industrial power unit with BN-800 reactor (its construction is under way at the Beloyarsk nuclear power plant site in Zarechny).

Successful functioning of SFR facilities formerly and currently operating in the Russian Federation, where elements of SFR technology were tested and upgraded, confirms a high level of industrial expertise in SFR technology[1]. The total operational lifetime of the SFR facilities reflects the scope of experience gained and is evidence of the level of expertise gained in this technology. Table 1 presents the data on the operational lifetime of various SFRs, both those operated in the past and those currently operating in the world[2]. The data presented in Table 2 give a clear picture of the experience gained on SFRs in different countries.

The history of domestic SFR development testifies to a regular and consecutive (stepwise) manner of gaining expertise in this reactor technology. Specific tasks were set for each milestone/phase; the goals of the next step or phase were determined with account taken of the results obtained at the previous step. The following major phases can be defined in implementing domestic SFR technology:

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[1] It should be noted that two pulsed fast reactors, IBR-1 (1960) and IBR-2 (1981), using sodium coolant were constructed at the Joint Institute for Nuclear Research at Dubna. However, they are not included in this review owing to their highly specific features.

[2] The operation time periods shown in Table 1 are calculated from the moment of ‘wet’ first criticality to the moment of final reactor shutdown. However, it should be noted that the period of facility decommissioning after final shutdown is also one of the operation phases and a certain operational experience is obtained at that time too.

TABLE 1. LIFETIME OF SFR OPERATIONS (as of 2009-10-31) [1, 2]

No.	Reactor (country)	First criticality date	Date of shutdown	Operational lifetime (a)
	Research facilities with SFR			
1.	EBR-I (USA)	1951-08-24	1963-12-30	12.4
2.	BR-5/BR-10 (Russian Federation)	1959-01-26[a]	2002-12-06	43.9
3.	DFR (UK)	1959-11-14	1977-03-23	17.4
4.	EBR-II (USA)	1963-11-11	1994-09	30.9
5.	EFFBR (USA)	1963-08	1972	9.3
6.	Rapsodie (France)	1967-01-28	1983-04-15	16.2
7.	SEFOR (USA)	1969	1972	~3
8.	BOR-60 (Russian Federation)	1969-12-14[b]		39.9
9.	JOYO (Japan)	1977-04-24		32.5
10.	KNK-II (Germany)	1977-10-10	1991-08-23	13.9
11.	FFTF (USA)	1980-02-09	1992-03-18	12.1
12.	FBTR (India)	1985-10-18		24.1
	Total (research facilities with SFR)			255.6
	Power generating reactor facilities with SFR			
1.	BN-350 (Russian Federation/Kazakhstan)	1972-11-29	1999-04-22[c]	26.4
2.	Phenix (France)	1973-08-31[d]		36.2
3.	PFR (UK)	1974-03-01	1994-03-31	20.1
4.	BN-600 (Russian Federation)	1980-02-26		29.7
5.	Super-Phenix (France)	1985-09-07	1998-02-02	12.4
6.	MONJU (Japan)	1994-04-05[e]		15.6
	Total (power reactor facilities with SFR)			140.4
	Total (research and power facilities with SFR)			396.0

[a] ‘Wet’ first criticality was on 1959-01-26 (‘dry’ first criticality was performed in July 1958).
[b] ‘Wet’ first criticality was on 1969-12-14 (‘dry’ first criticality procedures were in December 1968).
[c] The official date of shutdown is shown (in fact, the reactor was shut down in April 1998).
[d] Officially, this reactor was shut down on 2009-09-12. However, end-of-life tests on the critical facility are planned to be carried out until December 2009.
[e] This reactor is not in power operation after a sodium leak on 1995-12-08. However, it remains operational.

TABLE 2. LIFETIME OF SFR OPERATION IN DIFFERENT COUNTRIES (as of 2009-10-31)

No.	Country	Lifetime of research SFR operation (a)	Lifetime of power SFR operation (a)	Total lifetime of SFR operation (a)
1.	Russian Federation/ Kazakhstan	83.8 (32.8%)	56.1 (40%)	139.9 (35.3%)
2.	USA	67.7 (26.5%)		67.7 (17.1%)
3.	France	16.2 (6.4%)	48.6 (34.6%)	64.8 (16.4%)
4.	Japan	32.5 (12.7%)	15.6 (11.1%)	48.1 (12.1%)
5.	UK	17.4 (6.8%)	20.1 (14.3%)	37.5 (9.5%)
6.	India	24.1 (9.4%)		24.1 (6.1%)
7.	Germany	13.9 (5.4%)		13.9 (3.5%)
	All countries	255.6 (100%)	140.4 (100%)	396.0 (100%)

(a) Initial phase to justify SFR feasibility (based on BR-5 experience);
(b) Phase to confirm SFR viability at an industrial level (e.g. prototype BN-350 RF);
(c) Research and choice of principal design and technical proposals for a ‘demo’ industrial facility within the framework of development of the BN-600 RF design (considering experience gained through operation of BOR-60 and BN-350 facilities);
(d) Justification of reliability and safety of SFR technology (operation of industrial power unit with BN-600 reactor as an example).

The past two to three decades have been associated mainly with implementation of the last phase in the list above. However, all of the list items are closely interconnected. Therefore, if the analysis is limited to the last phase only, consecutive logics for developing SFR technology and the reasons for choosing a specific design and technical proposals will be lost. In this connection, the initial phases of domestic SFR evolution will be touched upon as well, to provide an historic background. In addition, this complex approach to the analysis of experience gained allows formulation of the tasks for the further phases. The following phases could be defined in accordance with further tasks of SFR technology development and upgrading:

(a) Demonstration of the closed fuel cycle of SFRs (on the basis of the BN-800 reactor);

(b) Commercialization phase of SFR technology (design development and construction of a commercial reactor (BN-K));
(c) Extended deployment phase of SFR technology (construction of a small series of commercial power units using the BN-K reactor).

2.1. Initial phases of development and mastery of SFR technology

After corroboration of the possibility of nuclear fuel breeding on research reactors BR-1 (1955, 100 W, air cooling) and BR-2 (1956, 100 kW, mercury coolant) at the Institute for Physics and Power Engineering (IPPE), a resolution for the construction of research fast reactor BR-5 with sodium coolant at the IPPE site was approved [3]. The main goals to be achieved (the challenges were successfully met by the BR-5 reactor) included the following:

(a) Experimental demonstration of feasibility of production and operation of SFRs with parameters corresponding to those of power facilities;
(b) Confirmation of the feasibility of practical implementation of fuel breeding in SFRs;
(c) Mastery of sodium technology and the basic elements of SFR technology within a separate facility;
(d) Fulfilment of tests with different fuel compositions and structural materials;
(e) Substantiation of fuel pin and fuel composition serviceability to acceptable burnup levels (maximum achieved burnup made 14.1% heavy atom for plutonium dioxide, 6.1% heavy atom for uranium monocarbide and 9% heavy atom for uranium mononitride).

The problems remaining to be resolved included the issue of development of a design for a reliable steam generator (SG). The SG with double wall heat exchange tubes operated at the BR-5 during the initial period was subsequently replaced by an air heat exchanger owing to leaks.

After the theory of SFR feasibility was proved experimentally, the task of confirmation of viability of this type of reactor facility at an industrial level was set. In this connection, designing the BN-350 RF was initiated in 1960, followed by the BN-600 RF (preliminary design development started in 1963). During these work periods, the need for experimental justification of materials and design and technical proposals chosen for the demonstration industrial facility appeared. This was the reason for construction of the BOR-60 test reactor (the decision to construct the BOR-60 was made on 1964-09-08). Therefore, the design and construction of the BOR-60, BN-350 and, later, BN-600 units were fulfilled in parallel for some of the time.

It is also necessary to note reconstruction of the BR-5 reactor into the BR-10 in 1971–1973, and then reconstruction in 1979–1983, which resulted in a considerable extension of experimental capacity of the reactor and enhanced its safety level. In subsequent periods of BR-5/BR-10 reactor operation, a series of investigations on mastering SFR technology elements were carried out:

(a) Experiments with the operation of failed fuel pins and substantiation of a cladding tightness monitoring system based on indication of delayed neutrons in the coolant;
(b) Detailed study of mass transfer and distribution of different impurities and nuclides (Mn, Co, Cs and others) over the primary circuit and development of methods for control of fission and corrosion products' activities in the coolant and on the walls of the primary circuit pipelines;
(c) Development of technology for the purification of coolant from oxides and radioactive caesium;
(d) Development of technologies for removal of non-drainable sodium residues from the circuit by vacuum distillation and washing (steam, steam–gas mixture) internal circuit surfaces of sodium followed by their decontamination, etc.

Currently, the BR-10 reactor is being prepared for decommissioning and is used for the development of technological processes proposed for SFR decommissioning. Extensive operational experience has been gained on the BR-5/BR-10 reactor, including that on sodium leaks, operation with failed fuel pins, determining lifetime of structural materials under irradiation and corrosion effects of sodium coolant, etc. This experience is used for the continued upgrading of SFR technology and improvement of its technical and economic performance.

The scheme for the BOR-60 RF models, in a full scope, included the design of the power plant with SFRs (a three circuit layout, with a possibility of heat removal towards the tertiary circuit and electricity generation). This gave an opportunity to test all basic elements (pumps, heat exchangers, SGs, cold traps, etc.) and reactor control systems, and implement research for SFR safety substantiation and upgrading sodium coolant technology. High neutron physical and thermal characteristics of the BOR-60 reactor make it possible to perform in-pile testing of fuel pins, fuel compositions and structural materials used for SFRs under conditions corresponding to those of power reactors. Thus, the BOR-60 has played an important role in the justification of various design and technical proposals for power SFRs and in adjusting different systems and technologies [4]. The following work should be mentioned, which contributed to SFR technology development and upgrading:

(a) Tests of various fuel compositions, structural materials and absorbent materials, including fuel pins with vibropacked MOX fuel;
(b) Tests of different types of SG, including sections of the modular BN-600 SG and the reverse SG (RSG)[3] of micromodular and modular design;
(c) Experiments to study the processes of water–sodium interaction under small (up to 0.2 g/s) and large (up to 0.25 kg/s) water leaks, checking various methods for leak indication and confirmation of serviceability and reliability of SG protection systems;
(d) Implementation of complex studies on the technology of sodium and sodium equipment, including justification of the serviceability of systems for sodium purification from impurities and from caesium, regeneration of oxide cold traps, decontamination of equipment in contact with sodium, etc.

Design operation time for the BOR-60 reactor ended in late2009 and now work is being carried out on extending its lifetime to 2015.

During operation of the BOR-60 reactor (more than 225 000 h in critical condition), considerable experience was gained on the faults of some reactor fuel elements, their inspection and repair, on the operation time achieved for non-replaceable reactor fuel elements under irradiation, on the lifetime achieved for main items of equipment, etc., which can be used in further upgrading SFR technology. Thus, for example, unique experience has been obtained with long term operation of the RSG without any leaks (29 year operation of RSG-1 and 19 year operation of RSG-2).

Figure 1 shows the distribution of unscheduled BOR-60 RF shutdowns by initial events and years for the period 1970–2008. Over the past 20 years, the main cause of the reactor shutdowns has been loss of the external power supply. It should be noted that none of the events shown in Fig. 1 ever resulted in any radiation consequences exceeding admissible safety levels.

The BN-350 RF operation on the whole has demonstrated the correctness of the principal solutions accepted in its design, demonstrated stability and ease of SFR control, and their sufficient reliability and safety. The problem of ensuring SG reliability arose during the initial period of the BN-350 unit's operation as a result of numerous leaks appearing due to inferior manufacturing quality of the heat exchange Fild's tubes used in SG evaporators (during the first two years

[3] Water–steam flows in the intertubular space; sodium is placed inside heat exchange tubes.

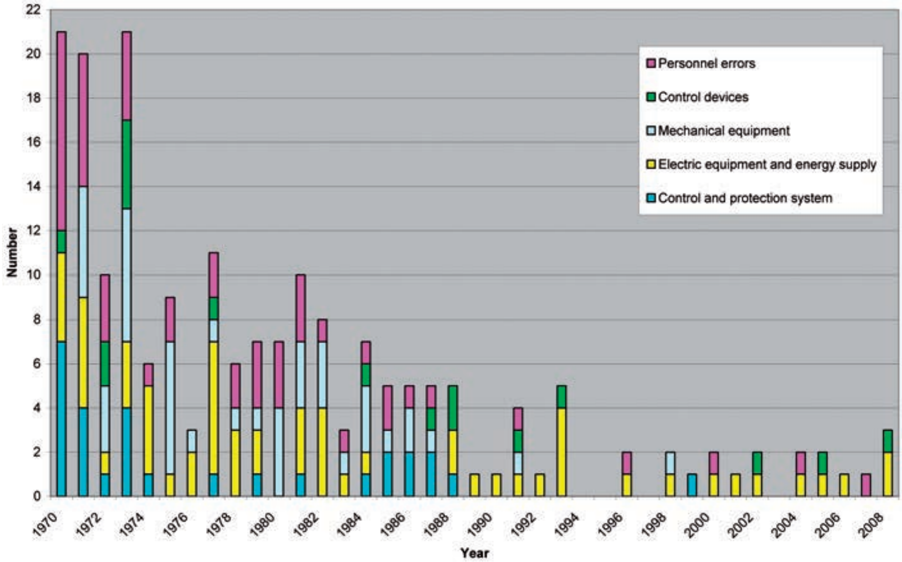

FIG. 1. Distribution of unscheduled BOR-60 shutdowns by initial events and years.

(1973–1975), eight leaks occurred, including three ‘large’ ones[4]; twelve leaks altogether occurred in the SG with Fild’s tubes over the entire operation period [5]). The fast, self-development of small SG leaks into large ones was revealed. This required improvement of SG protection systems against leaks and increasing their response speed. The experience concerning leaks in the BN-350 SG subsequently influenced the transition from vessel-type SGs (BN-350 SG with Fild’s tubes, BOR-60 vessel coil-type SG) to sectional–modular SGs (BN-600 and BN-800 sectional–modular SGs, BN-350 micromodular SG “Nadezhnost”, BOR-60 micromodular and modular SG).

Summarizing BN-350 operational experience, it can be said that it has confirmed the possibility of implementing this reactor technology at the level of an industrial power unit. The knowledge gained during BN-350 operation has provided a reliable basis for development and upgrading BN-600 and BN-800 designs, designs of advanced SFRs and for further upgrading SFR technology.

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[4] The intercircuit SG leaks are subdivided into small and large leaks. The small leak, in contrast to a large leak, is not accompanied by a change of the basic parameters of the secondary circuit (sodium pressure and gas pressure in expansion tank, coolant flowrate in the circuit).

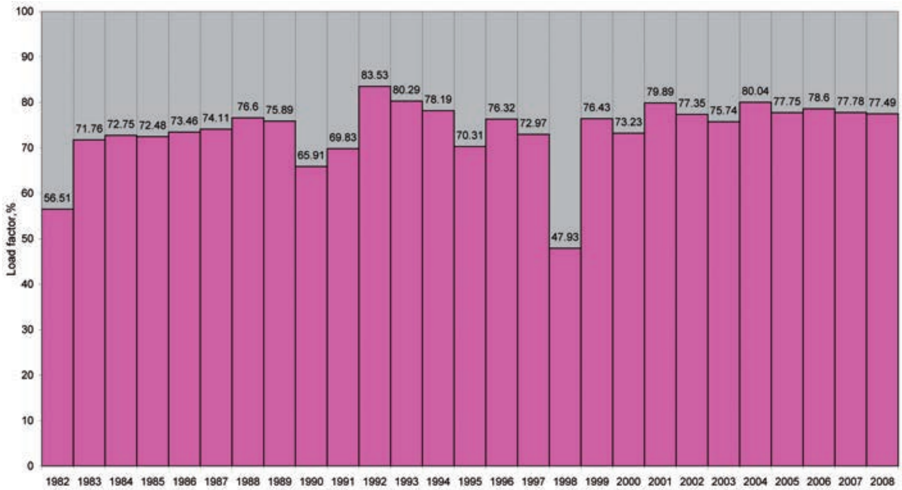

FIG. 2. Change of load factor during BN-600 power unit's commercial operation.

2.2. Phase of justification of reliability and safety of SFR technology

The experience gained on the BR-5/BR-10, BOR-60 and BN-350 units was taken into consideration in the development of the BN-600 RF design and choice of design and technical proposals. The principal difference of the BN-600 from previous SFR designs is the integral configuration of the primary circuit. The correctness of solutions accepted in the BN-600 design was confirmed subsequently by its successful operation for nearly 30 years. The BN-600 was connected to the grid on 8 April 1980; design power level was reached in December 1981. Since 1982, power unit No.3 of Beloyarsk nuclear power plant (with a BN-600 reactor) has operated as a commercial power unit. Figure 2 shows the change of load factor for power unit No.3 of the Beloyarsk nuclear power plant during commercial operation. The average load factor during this period (1982–2008) is 73.82%.

For the whole period of its operation (about 205 000 h in critical condition), the BN-600 produced more than 110 billion kW·h of electrical power, which was a considerable contribution to the power supply of the Urals region from one of the most cost effective and ecologically friendly power units. Thus, release of gaseous radioactive products into the atmosphere, as a rule, did not exceed 1% of the admissible level. The amount of solid and liquid radwaste has been minimal as well, not exceeding 50 m^3 annually. Dose burdens on the personnel are below the average values for the nuclear industry as a whole.

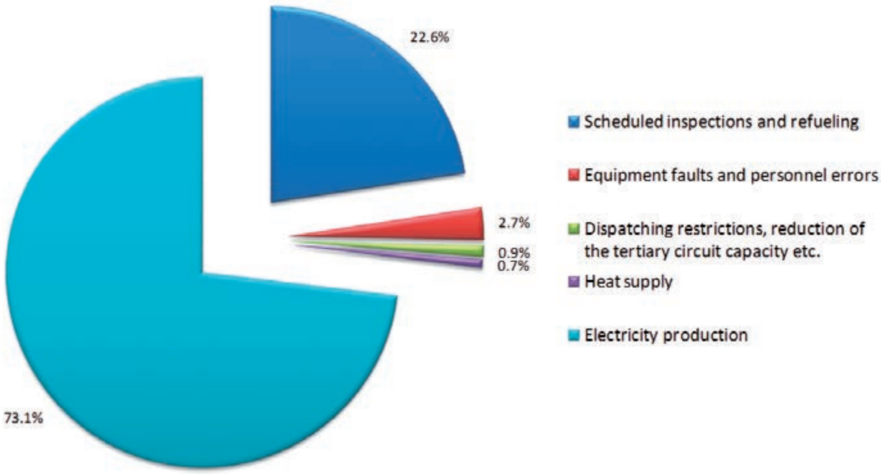

FIG. 3. Distribution of the causes of the decrease in the BN-600 power unit's load factor.

Figure 3 shows the results of detailed analysis of the causes of the decrease in the BN-600 load factor for the period 1982–2004 [6]. The scheduled value of the load factor loss is caused by the duration of power unit shutdown periods for carrying out scheduled preventive repair and reactor refuelling. Now, the duration of the annual reactor shutdown period for scheduled preventive repair is mainly determined by the rated time of complete overhaul of turbine generators (50 d) and the time required for reactor refuelling (twice a year). In recent years, the average scheduled preventive repair duration has been about 71 d[5].

Operational experience gained with the BN-600 power unit in recent years (upon mastering SFR technology and adjusting operation of the main components) shows that unscheduled losses of load factor are mainly caused by failures of the components of the tertiary circuit and power supply system, and are equal, on average, to 1.1% annually. Figure 4 shows the time distribution of equipment faults and personnel errors that took place with the BN-600 power unit in 1982–2008.

The following work has been undertaken at the BN-600 reactor during this period:

(a) Long duration tests of large-sized sodium components;
(b) Mastering sodium technology on an industrial scale;

[5] In the past two years, scheduled preventive repair had longer durations caused by replacement of SG modules and implementation of other measures for lifetime extension of the BN-600 RF.

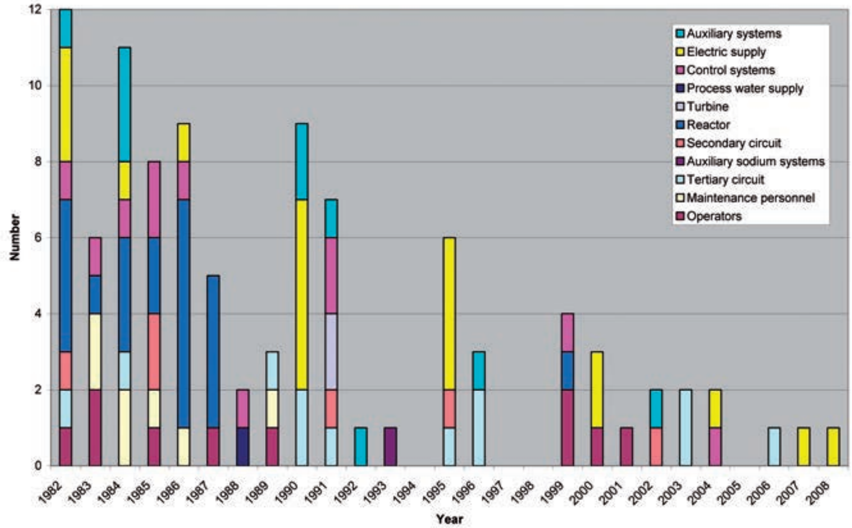

FIG. 4. Time distribution of equipment faults and personnel errors during the BN-600 power unit's commercial operation.

(c) Development and optimization of operation modes;
(d) Achievement of acceptable levels of fuel burnup;
(e) Mastering technology of replacement and repair of sodium components, including the main equipment (pumps, SGs, intermediate heat exchangers, rotating plugs).

A programme of steadily increasing the design level of uranium oxide fuel burnup is carried out at the BN-600 reactor. In the course of the first modernization of the BN-600 core (transition to 01M variant), the reactor core and fuel pin configurations were changed with a decrease of fuel pin linear heat rating and the FSA reloading scheme was optimized. A further increase of fuel burnup level was achieved by replacing the structural materials of fuel pin cladding and the FSA wrapper (modification 01M1). Successful operation of the BN-600 reactor with the 01M1 core and the R&D cycle performed made it possible to increase the design level of fuel burnup to 11.1% heavy atom, and to change over to a longer fuel operating life in the reactor with fourfold reloading (modification 01M2) [7, 8]. The main design characteristics of all BN-600 reactor core modifications which changed during these modernizations are shown in Table 3.

The fact that design values of operation time and the lifetime of large-sized sodium equipment have been achieved and even surpassed is one of major results obtained during BN-600 operation. Table 4 shows the data on parameters

TABLE 3. BN-600 REACTOR CORE DESIGN EVOLUTION

Parameter	Reactor core type			
	01	01M	01M1	01M2
Period of the core type operation	1980–1986	1987–1991	1993–2004	since 2005
Active core height (mm)	750	1000	1030	1030
Axial blankets height (mm):				
— Upper	400	300	300	300
— Lower	400	380	350	350
Number of fuel enrichment zones	2	3	3	3
Fuel pin gas plenum length (mm)	808	653	653	653
Core structural materials:				
— Fuel pin cladding[a]	EI-847	EI-847	ChS-68cw	ChS-68cw
— FSA wrapper	Cr16Ni11Mo3	Cr16Ni11Mo3Ti	EP-450	EP-450
Fuel pin maximum linear heat rating (kW/m)	54.0	47.2	≤48.0	≤48.0
Maximum fuel burnup (% heavy atom)	7.2	8.3	10	11.1
Maximum radiation dose to cladding (dpa)	43.5	53.9	75.0	82.0
Fuel operating life (fpd)	200/300	300/495	480	560/720
Core fuel cycle (fpd)	100	165	160	120/160[b]
Fuel inventory in core (kg)	8260	11630	12090	12090
Average fuel burnup (MW·d/kg U)	42.5	44.5	60.0	70.0

[a] EI-847 (Cr16Ni15Mo3Nb) — austenitic steel, ChS-68cw (Cr16Ni15Mo2Mn2TiB) — austenitic steel, EP-450 (Cr12MoBnVB) — ferritic-martensitic steel.

[b] 120 fpd — summer fuel cycle, 160 fpd — winter fuel cycle.

achieved during domestic operation of the BN-600 and other facilities with SFRs. The data presented testify to good compatibility of sodium coolant with the structural materials used and its low corrosion activity over the mastered range of SFR parameters.

Replacement of the following main equipment of the BN-600 power unit was performed during its operation:

TABLE 4. ACHIEVED PARAMETERS FOR OPERATION TIME AND LIFETIME OF SFR EQUIPMENT WITHOUT OVERHAUL

Type of equipment	BR-5/BR-10 (h)	BOR-60 (h)	BN-350 (h)	BN-600 (h)
Non-replaceable equipment:				
— Reactor vessel	150 000	225 000	170 000	205 000
— Primary piping	300 000	225 000	170 000	205 000
Sodium pumps	Electromagnetic 170 000	Mechanical 260 000	Mechanical 100 000	Mechanical 105 000
Intermediate heat exchangers	300 000	225 000	170 000	205 000
SGs		RSG 155 000	150 000	Evaporators 125 000

(a) Four sets of primary sodium main circulating pumps;
(b) One set of secondary sodium main circulating pumps;
(c) One set of mechanisms for the control and protection system;
(d) Three sets of guide tubes for the control and protection system rods;
(e) A nearly complete set of SG modules plus one set of evaporator SG modules;
(f) One intermediate heat exchanger.

Unique repair of the small rotating plug has been fulfilled.

In the initial phase of BN-600 operation (while expertise was being acquired by personnel on sodium technology, design and technical proposals were verified and refined, operating modes were adjusted and equipment manufacturing defects were detected), experience related to outside sodium leaks and intercircuit leaks in the SGs was gained. Altogether, there were 27 external sodium leaks and 12 leaks in SGs during BN-600 operation [9].

All 27 external sodium leaks were detected in due time by detection systems or by operating personnel. Powders were used for localizing and extinguishing non-radioactive sodium fires. There was only one case of a radioactive sodium leak from the auxiliary pipeline of the primary circuit and the design algorithm for confinement of sodium fires and for limiting their consequences was implemented successfully. In this case, radioactivity release (10.7 Ci) was much lower than the permissible limit. The experience gained on sodium leaks demonstrated sufficient efficiency of the protection systems for localizing their consequences.

Sectional–modular SGs used in the BN-600 power unit have demonstrated high performance for the whole period of operation. Half of the 12 leaks of vapour/water into sodium mentioned above occurred in the first year of operation; they were caused by development of latent manufacture defects. Inter-circuit leaks occurred mainly in the superheaters (six events) and reheaters (five events), whereas only one leak occurred in the evaporator. All SG leaks were suppressed by regular means and did not result in any emergencies [10].

In evaluating all abnormal events that occurred during BN-600 operation, including those associated with sodium leaks, it should be emphasized that none of them ever resulted in radiation impact on the population or the environment; all of them were below the International Nuclear Event Scale range set by the 'off-site impact' criterion, i.e. they were insignificant.

To date, the BN-600 power unit has already operated for 30 years, its orginal design lifetime. The following facts should be mentioned in this connection:

(a) The last external sodium leak occurred at the BN-600 more than sixteen years ago, in May 1994.
(b) As for leaks in SGs, during the past 24 years of BN-600 operation there has only been one small leak, in January 1991. The SGs have operated without any intercircuit leaks for nearly 19 years, in spite of numerous replacements of SG modules made in this period according to the regulations and to SG lifetime.
(c) As mentioned above, faults that occurred in recent years were mainly related to the technological equipment of the tertiary circuit and electrical power supply systems, not to sodium systems.

Thus, the BN-600 reactor has demonstrated high safety and reliability indices during its commercial operation and, therefore, has allowed a successful solution of the task set, i.e. justification of the safety and reliability of SFR technology at an industrial level, in particular, sodium coolant technology.

Successful operation of the BN-600 power unit served as the basis for organizing activities to extend its design lifetime from 30 years to 45 years. These activities have been carried out in the following areas:

(a) Justification of serviceability of non-replaceable elements of the reactor plant for an additional period of operation;
(b) Inspection and lifetime extension of elements not planned to be replaced;
(c) Replacement of equipment;
(d) Implementation of stipulated measures on enhancement of power unit's safety;

(e) Development of the Report on Profound Safety Estimation of the power unit and a complete set of substantiation documentation for issue of a new licence on operation.

Among the most important actions implemented within the frame of these activities planned to be completed in 2010, the following should be pointed out:

(a) Confirming serviceability of non-replaceable elements for a 45 year lifetime;
(b) Equipping the power unit with the second complete emergency protection system and redundant control room;
(c) Installing an additional decay heat removal system using an air heat exchanger;
(d) Replacing SG modules (49 of 72 modules have been replaced);
(e) Increasing the seismic stability of the power unit's systems and equipment.

Thus, extension of the BN-600 power unit lifetime to 2025, as anticipated, would promote, first, further mastery of SFR technology, and second, its further development via training of the personnel for new power units with SFRs.

3. PROSPECTS FOR SFR DEVELOPMENT IN THE RUSSIAN FEDERATION AND FUTURE CHALLENGES

Successful experience with the BN-600 operation has created good prerequisites for the further development of SFRs in the Russian Federation. Currently, power unit No. 4 of the BN-800 reactor is under construction at the Beloyarsk nuclear power plant, with commissioning scheduled for 2012 (see Fig. 5). The design of an advanced large-sized commercial SFR (BN-K) is under development.

One of the major tasks to be resolved in the course of BN-800 operation is demonstration of the closed nuclear fuel cycle. The implementation of the closed nuclear fuel cycle using SFRs will mean mastering fully the complexity of SFR technology. This will provide an opportunity to resolve the problems of extension of the fuel base for the nuclear power industry and utilization of spent nuclear fuel, including minor actinides.

In order to resolve the tasks of nuclear power listed above, the Federal Target Programme on Nuclear Power Technologies of a New Generation for the Period 2010–2015 with the Outlook to 2020 has been developed. In accordance with this programme, further development of the SFR is stipulated within the

(a)

(b)

FIG. 5. Aerial view of the BN-800 reactor vault (a) and assembly of a set of the BN-800 reactor vessel bottoms (b).

framework for creating a new technological platform for nuclear power based on transition to the closed nuclear fuel cycle using 4th generation fast reactors.

Development of the advanced commercial BN-K reactor design with an installed electrical power of 1200 MW (BN-1200) is planned as a 4th generation SFR. It is supposed that this design will ensure economic characteristics comparable with those of a thermal nuclear power plant and achieve a safety level that meets the requirements defined for the 4th generation reactors. Thus, design and construction of the BN-1200 reactor will be carried out within the SFR technology commercialization phase. Possibilities for construction of a pilot nuclear power plant with a BN-1200 reactor by 2020 and a small series of BN-1200 units by 2030 that will correspond to the phase of deployment of SFR technology are under discussion.

In order to provide R&D necessary for development of the 4th generation innovation fast reactors, the Federal Target Programme stipulates modernization of the related experimental basis. In particular, construction of a multifunctional research fast reactor using sodium coolant is scheduled after 2015; it will have a wide range of experimental loops and channels.

Development of the complex of technologies for SFR decommissioning is considered one of the significant tasks for ensuring further successful development of SFR technology. The implementation of this task is stipulated on the basis of the BR-10 reactor.

4. CONCLUSIONS

The review of SFR development in the Russian Federation presented in this paper gives evidence of their systematic progressive evolution during the whole period of activities in this area (more than 50 years).

The analysis of experience gained domestically on the SFR in the past 30 years, primarily based on the results of the successful and stable operation of power unit No.3 of the Beloyarsk nuclear power plant with the BN-600 reactor, allows a conclusion to be drawn about industrial mastery of SFR technology, in particular, sodium technology. The SFR operating indices achieved provide good prerequisites for their further commercialization and enhancement of safety.

The Federal Target Programme on Nuclear Power Technologies of a New Generation for the Period 2010–2015 with the Outlook to 2020 suggests further domestic development of SFRs in the framework of creating a new technological platform for nuclear power based on the transition to the closed nuclear fuel cycle with 4th generation fast reactors.

ACKNOWLEDGEMENTS

The authors express their gratitude to all Russian scientists, engineers, design and operating personnel who have contributed to the creation and development of SFR technology and the accumulation of unique experience in this area.

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ADVANCED SIMULATION FOR FAST REACTOR DESIGN*

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

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ISSUES AND CHALLENGES OF FAST REACTORS: IMAGINATIVE BREAKTHROUGH VERSUS BUSINESS AS USUAL

M. SALVATORES
Commissariat à l’énergie atomique,
Cadarache, France

1. INTRODUCTION

At an historical meeting on 26 April 1944, Enrico Fermi opened the way towards fast reactors with a genius’ intuition of the potential neutron surplus in the neutron balance of a fission reactor core where the neutrons would not have been thermalized:

DISCUSSION ON BREEDING

Excerpt from Report N-1729. (Notes on meeting of April 26, 1944).

Present: FERMI, ALLISON, SZILARD, WIGNER, WEINBERG, SEITZ, MORRISON, COOPER, VERNON, TOLMAN, WATSON, OHLINGER.

This leads to the equation

$$\text{(1)} \qquad \text{Breeding ratio (without losses)} = \frac{\nu_9 + \beta\nu_8 - 1 - \alpha - \beta}{1 + \alpha}.$$

What value of this ratio can we hope to obtain in a “ fast ” pile (with a core of 9 surrounded by 8) ?

The most important term in (1) is $\nu_9 - \alpha - 1$. It can be evaluated essentially by a single experiment:

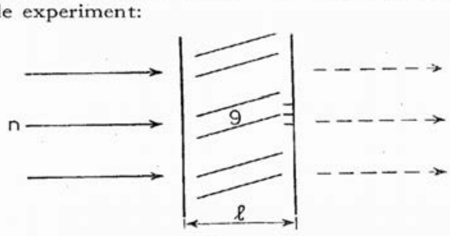

Fig. 2.

The neutron beam traversing a layer l cm thick of 9, is decreased by absorption and swelled by fission neutrons. The intensity of the transmitted beam is changed by a factor of

$$1 + \sigma_f l (\nu_9 - 1) - \sigma_f \alpha l$$

or

$$1 + l [\sigma_f (\nu_9 - \alpha - 1)].$$

A simple transmission experiment thus permits to evaluate the expression in square brackets. The result of such an experiment, performed at Y is

$$\nu_9 - \alpha - 1 = 1.85$$

These short notes are the foundation of the discovery of fast reactor flexibility (i.e. to breed fissile material, to burn radioactive wastes or even to breed while burning!). In summary, whenever an option is considered for sustainability and optimized waste management, the initial Fermi intuition is confirmed and a fast neutron spectrum is consistently required.

2. A MAJOR, MEDIUM/SHORT TERM CHALLENGE

At present, there is a wide convergence on the choice of sodium as coolant, with oxide or metal (e.g. for high conversion ratio) fuel. However, several innovative features have been discussed, which should be verified experimentally at the level of a prototype representative of successive industrial realizations. Moreover, irradiation capabilities are, and will remain for some time, scarce. Finally, it also seems sensible to explore/develop a viable backup option, such as lead (or lead–bismuth) coolant and nitride fuel, or gas coolant and carbide fuel.

In this context, both (i) an innovative sodium cooled prototype and (ii) an experimental reactor for exploring a backup option should/could be joint international initiatives.

3. MEDIUM TERM CHALLENGES

After a long period of uncertainty and doubts, it seems today that the potential mass deployment of fast reactors could be related to the successful solution in the medium term of a number of significant challenges:

(a) High availability and reliability (a major utility requirement).
(b) Reversibility (from burner to breeder and vice versa, in order to prepare for new technology breakthroughs or new needs).
(c) Convergence of safety approaches at the international level. In this context, the recriticality issue is to be revisited.
(d) Fuel and cladding performance, in order to foresee core performances beyond 20–30 at.% burnup and beyond 200 dpa. The transient behaviour of minor actinide loaded fuel is an issue not explored up to now but one that is crucial for the development of such fuels.
(e) Plant simplification (e.g. by reassessing the options for the intermediate circuit) and cost reduction.
(f) Feasibility of cores with a conversion ratio higher than 1.5 while accounting for non-proliferation concerns.

(g) Reduction of design uncertainties in all fields with the use of advanced simulation and innovative validation experiments.

It is important to realize that preliminary answers are available for each of these challenges. Some examples are given below:

(a) *Availability*: The BN-600 has already shown a remarkable record (availability largely above 70%), which has to be compared to the progressive increase of the load factor for LWRs during the last three decades.
(b) *Reversibility*: An early demonstration was successfully attempted in the frame of the CEA's CAPRA project in the 1980s.
(c) *Recriticality*: The EAGLE experiment has already produced significant results and more are expected in support of the innovative FAIDUS S/A concept of the Japan Atomic Energy Agency.
(d) *Fuel and cladding performance*: New oxide dispersion strengthened steel and fuel fabrication routes have been the subject of experimental demonstrations, particularly in Japan. As for the presence of minor actinides in the fuel, as foreseen by current transuranic management strategies, the international GACID experiment (minor actinide loading in an oxide fuel) is planned within a trilateral collaboration (France, Japan, United States of America) in the framework of the Generation IV International Forum and the CRIEPI/ITU METAPHIX experiment (metal fuel with minor actinide loading), which has been performed in the PHENIX reactor and a follow-up activity (irradiated fuel processing with successive new fuel fabrication) is expected. The availability of the TREAT reactor in Idaho is being discussed in order to make available to the international community a unique tool for the validation of innovative fuels under transient conditions.
(e) *Plant simplification and cost reduction*: The JSFR cost evaluation and further recent estimations both in India and the Russian Federation seem to indicate potential significant capital cost reduction.
(f) *Reduction of uncertainties*: Advanced simulation and validation experiments have the potential to reduce design uncertainties significantly, under the condition of keeping available or even upgrading experimental facilities in the fields of neutronics, thermohydraulics, and material and fuel assessment.

However, there are also long term challenges that can help to make even more fundamental breakthroughs in order to consolidate nuclear energy deployment on a very large scale later next century. A personal list includes:

(a) The need to revisit the standard choice of a solid fuel, to envisage a different approach to recriticality and to increase burnup.
(b) Simplified fuel cycles and waste management strategies.
(c) The reprocessing options (both aqueous and pyrometallurgical) are mostly associated with the choice of a solid fuel. Different choices could open the way to alternative fuel cycles.
(d) The potential for ultra-long life cores or their present conceptual development (e.g. travelling wave reactors) deserve closer scrutiny in the context of international expert initiatives.
(e) Innovative materials development ('gateway' towards much higher temperatures and burnup).
(f) Advanced simulation for future fuel design is also a potentially radical development and improvement on current practices.

In this framework, fast reactors offer a wide range of possible transformational concepts for both the reactor and the associated fuel cycles. Stay 'tuned' and prepare for a celebration in… 2044!

4. FINAL REMARKS

This is an exciting time because:

(a) The Monju restart is planned very soon.
(b) New constructions (CEFR, PFBR, BN-800) are close to completion.
(c) New strategic requirements for fast reactor mission.
(d) Emergence of regional visions.

The present design, construction and operational experience will expand dramatically and we can foresee both new international initiatives and the beginning of a new phase of fast reactor development leading towards mass commercialization. In this context, it is not fully clear how competition and enhanced international cooperation will coexist, and novel frameworks will be needed.

Focused R&D activities will still dominate the scene for the next 10–20 years and a 'business as usual' approach is not a guarantee of success. 'Imaginative breakthroughs' will be needed to innovate and to cope with the most crucial issues.

International cooperation will be essential:

(a) To share experimental facilities;
(b) To agree and converge on a safety approach at the international level;
(c) To provide cutting-edge opportunities for education and training.

Hopefully, the revival represented by this conference will be confirmed by an expansion of innovative ideas and their realization. The key will be to stay focused on crucial issues; as Hotei-san reminds us, “focus on the moon and not on the finger pointing to the moon…”

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