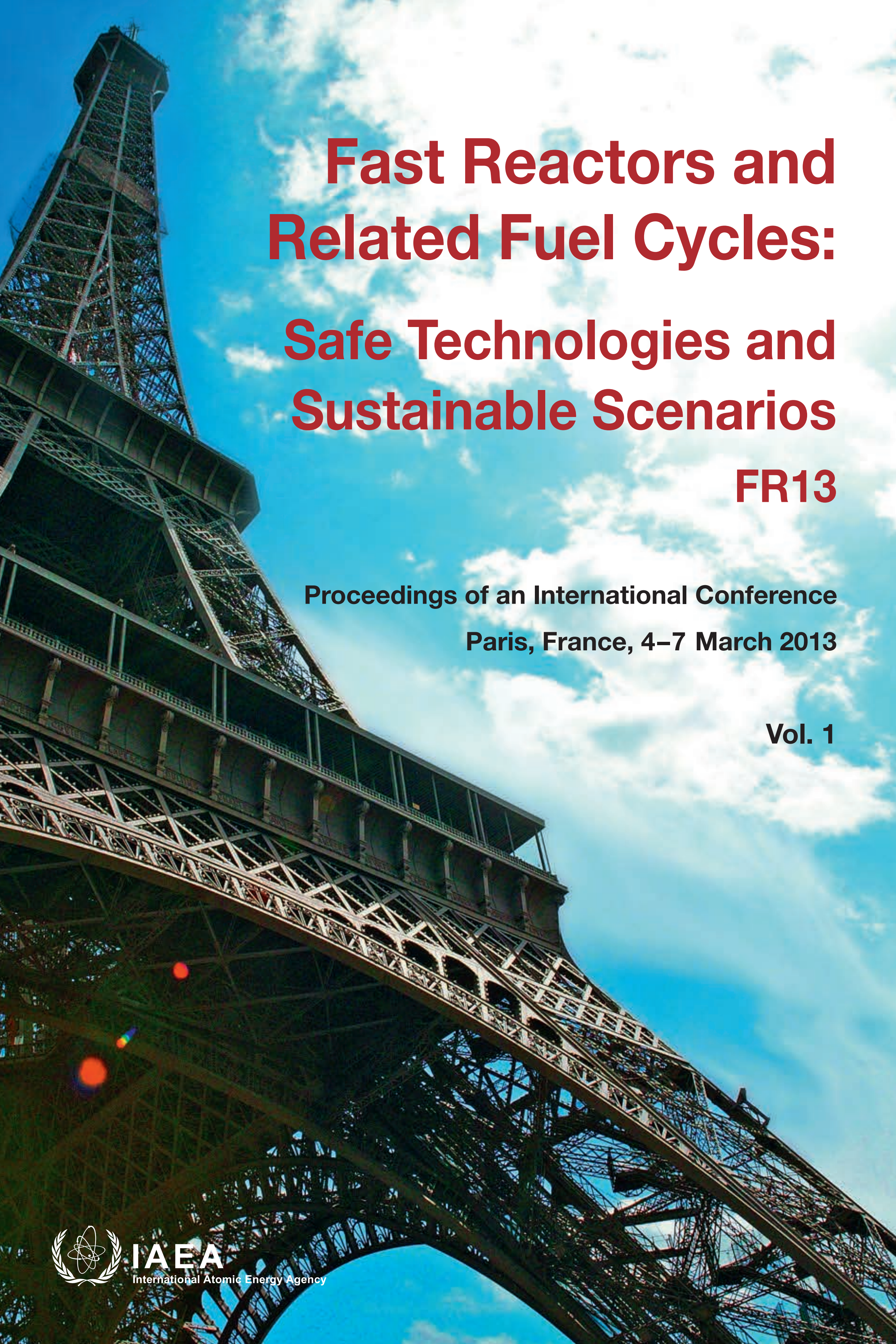


Fast Reactors and Related Fuel Cycles:

Safe Technologies and Sustainable Scenarios

FR13

Proceedings of an International Conference

Paris, France, 4–7 March 2013

Vol. 1

IAEA
International Atomic Energy Agency

FAST REACTORS AND
RELATED FUEL CYCLES:
SAFE TECHNOLOGIES AND
SUSTAINABLE SCENARIOS
(FR13)

VOLUME 1

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''.

PROCEEDINGS SERIES

FAST REACTORS AND RELATED FUEL CYCLES: SAFE TECHNOLOGIES AND SUSTAINABLE SCENARIOS (FR13)

EDITED BY S. MONTI

PROCEEDINGS OF AN INTERNATIONAL CONFERENCE ON
FAST REACTORS AND RELATED FUEL CYCLES:
SAFE TECHNOLOGIES AND SUSTAINABLE SCENARIOS (FR13)
ORGANIZED BY THE INTERNATIONAL ATOMIC ENERGY AGENCY,
HOSTED BY THE GOVERNMENT OF FRANCE THROUGH
THE FRENCH ALTERNATIVE ENERGIES AND
ATOMIC ENERGY COMMISSION AND
THE FRENCH NUCLEAR ENERGY SOCIETY
IN COOPERATION WITH THE
OECD NUCLEAR ENERGY AGENCY AND
HELD IN PARIS, 4–7 MARCH 2013

In two volumes

VOLUME 1

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2015

Marketing and Sales Unit, Publishing Section
International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria
fax: +43 1 2600 29302
tel.: +43 1 2600 22417
email: sales.publications@iaea.org
http://www.iaea.org/books

Printed by the IAEA in Austria
April 2015
STI/PUB/1665

IAEA Library Cataloguing in Publication Data

Fast reactors and related fuel cycles : safe technologies and sustainable scenarios : FR13 : proceedings of an International Conference held in Paris, France, 4-7 March 2013 — Vienna : International Atomic Energy Agency, 2015.
2 vol. ; 24 cm. — (Proceedings series (International Atomic Energy Agency), ISSN 0074–1884) STI/PUB/1665
ISBN 978–92–0–104114–2
Includes bibliographical references.

1. Fast reactors — Congresses. 2. Fast reactors — Safetymeasures — Congresses. 3. Nuclear fuels — Management — Congresses. 4. Nuclear reactors — Technological innovations — Congresses. I. International Atomic Energy Agency.

IAEAL 15–00967

FOREWORD

The potentialities of fast neutron reactors and closed fuel cycles have been recognized since the earliest days of nuclear energy, dating back to the 1950s. With the achievable breeding ratio and the multiple recycling of the fissile materials obtained from the spent fuel, fast reactors allow full utilization of the energy potential of natural resources, namely uranium and thorium. In this way, the sustainability of nuclear power is enhanced in terms of resource preservation and management of high level and long lived radioactive waste, which is reduced in volume, radiotoxicity and heat load.

Despite the accident that occurred at the Fukushima Daiichi nuclear power plant in 2011, nuclear power remains an important option for many countries to ensure energy security and address growing energy needs and public concern about the environment. In this context, the development of innovative nuclear energy systems, in particular fast neutron systems and related closed fuel cycles, is widely considered a fundamental step for ensuring the long term sustainability of nuclear energy.

For almost fifty years, the IAEA has been supporting the development and deployment of fast reactor technology, serving interested Member States as an important forum for fast reactor information exchange and collaborative research and technology development. Since 1967, the keystone of the IAEA's efforts in this field has been the Technical Working Group on Fast Reactors (TWG-FR), a group of experts providing advice and support for programme implementation, reflecting a global network of excellence and expertise in the areas of advanced technologies and R&D for fast reactors. The TWG-FR coordinates its activities with other IAEA projects, especially those of the Technical Working Group on Nuclear Fuel Cycle Options (TWGNFCO), the Department of Nuclear Sciences and Applications, the Department of Nuclear Safety and Security, and the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO).

Among the wide range of activities and initiatives, the International Conference on Fast Reactors and Related Fuel Cycles is one of the most important events. The previous International Conference on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), held on 7–11 December 2009 in Kyoto, Japan, was attended by a large number of participants and produced favourable results. Four years later, on 4–7 March 2013, the fast reactors community gathered in Paris for the International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13). The conference was attended by almost 700 experts from 27 countries and 4 international organizations representing different fields of fast reactor and related fuel cycle technology.

The success of FR13 catalysed further collaboration and alliances for fast reactor development programmes. These Proceedings are accompanied by a CD-ROM of contributed papers.

The IAEA would like to express its appreciation to the Government of France for hosting the conference through the French Alternative Energies and Atomic Energy Commission (CEA) and the French Nuclear Energy Society (SFEN), to the OECD Nuclear Energy Agency, to the members of the International Advisory Committee, the International Scientific Programme Committee, the Local Organizational Committee and the Secretariat of the Conference for the commitment shown in organizing and convening the conference.

The IAEA officers responsible for this publication were Mr. S. Monti and Mr. U. Basak of the Department of Nuclear Energy.

CONTENTS

TECHNICAL TRACK 4:

FAST REACTOR MATERIALS:
ACHIEVEMENTS AND NEW CHALLENGES

SUMMARY

INTRODUCTION

The accident which occurred at the Fukushima Daiichi nuclear power plant in 2011 drew new attention of public opinion to the present and future roles of nuclear power in the world. Two years after this event, although many countries are discussing the prospects for nuclear energy, there is still a general conviction that nuclear energy will continue to represent an important resource for the future, motivated by the recognition of its potential to successfully address the growing energy needs and the concerns for the environment.

The development of fast neutron systems and related closed fuel cycles represents a necessary step for assuring long term sustainability of nuclear energy utilization. Fast reactors operated in closed fuel cycle are potentially able to provide energy for the next thousand years with the uranium resources already known and to minimize the volume, the heat load and the lifetime of the most hazardous nuclear waste. However, to achieve the full potential of fast neutron reactors, reaching a real breakthrough in the utilization of nuclear energy, the research and technology developments will have to demonstrate the fulfilment of modern criteria of economic competitiveness, stringent safety and security requirements, sustainable development, proliferation resistance and public acceptance.

On the basis of the extensive technology base gained through the past operational experience with experimental, prototype and commercial sodium cooled fast reactors (SFRs), several projects are currently under way, and strategies and technological pathways for innovative fast neutron systems are clearly identified. With respect to current projects, the commercial size BN-600 MW(e) SFR has been in operation in the Russian Federation for more than thirty years and the China Experimental Fast Reactor was connected to the electric grid in 2011. Furthermore, construction work on prototype fast breeder reactors, the PFBR in India and the BN-800 in the Russian Federation, is in progress. Besides these projects, a number of countries are carrying out intense research programmes and developing innovative fast neutron systems, for which an understanding and the assessment of different design options and related safety characteristics, based on the present knowledge as well as on new scientific and technological research efforts, is of paramount importance.

In this context, international cooperation plays a key role in accelerating and catalyzing technology advance. It was in this light that the IAEA organized in 2009 the first International Conference on Fast Reactors and Related Fuel Cycles — FR09, held in Kyoto, Japan, on 7–11 December, in order to respond to the increasing worldwide interest on fast reactor technology. Four years later, following the successful experience of FR09, almost 700 experts from 27 countries and 4 international organizations gathered in Paris on 4–7 March 2013 for the International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios — FR13, organized by the IAEA and hosted by the Government of France through the French Alternative Energies and Atomic Energy Commission (CEA) and the French Nuclear Energy Society, in cooperation with the OECD Nuclear Energy Agency (NEA).

The conference was structured so that it covered all the major technical aspects of fast reactors and related fuel cycles. The objective of the conference was to provide a forum for the exchange of information on fast reactor and fuel cycle technology advances, and related safety, economic and proliferation resistance issues. The aim of the conference was also to identify gaps and key issues that need to be addressed towards the industrial scale introduction of fast reactors, including public acceptance. Several existing fast reactors, current construction projects, innovative fast reactors concepts that are under development at national and international levels were reviewed and discussed.

The conference was opened with an opening session and a plenary session in which national and international programmes on fast reactors and related fuel cycles were presented. Major advances in several key areas of technological development were presented through 208 oral presentations made over 41 technical sessions, which were organized within ten topical tracks:

— Technical Track 1: Fast Reactor Designs: Goals and Paths of Progress;
— Technical Track 2: Fast Reactor Technologies, Components and Instrumentation;
— Technical Track 3: Fast Reactor Safety: Post-Fukushima Lessons and Goals for Next Generation Reactors;
— Technical Track 4: Fast Reactor Materials: Achievements and New Challenges;
— Technical Track 5: Fast Reactor Fuels and Transmutation Targets: Development and Irradiation Experiments;
— Technical Track 6: Fast Reactor Fuel Cycle: Processes and Demonstrations, Including Partitioning and Transmutation;
— Technical Track 7: Experimental Tests, Data and Advanced Simulation;
— Technical Track 8: Fast Reactor Deployment, Scenarios and Economics;
— Technical Track 9: Fast Reactor Operation and Decommissioning: International Experience;
— Technical Track 10: Skill Capabilities, Professional Development, Knowledge Management.

Additionally, 157 posters complemented the overall picture of the scientific and technical developments under way worldwide.

It is worth noting that all the 365 papers submitted to FR13 were peer reviewed by the members of the International Scientific Programme Committee and then revised by the authors.

The conference included also two panel events devoted to safety design criteria for fast reactors and sustainability of advanced fuel cycles. A Young Generation Event dedicated to young professionals involved in fast reactor programmes and projects was also organized as a plenary session. A closing session, chaired by C. Behar, Director of Nuclear Energy at the French Alternative Energies and Atomic Energy Commission (CEA) and Chairman of the conference and S. Monti from the IAEA, reviewed and summarized the main outcomes of the conference.

OPENING SESSION

The Opening Session, chaired by Mr. C. Behar (CEA) and Mr. S. Monti (IAEA), comprised four welcome addresses and opening remarks provided by Mr. L. Michel, Director General of the French Ministry of Ecology, Sustainable Development and Energy, the IAEA Director General Mr. Y. Amano, Mr. B. Bigot, CEA Chairman, and Mr. Y. Sagayama, Former Chairman of Generation IV International Forum (GIF).

The speakers welcomed the conference participants and expressed their appreciation and gratitude to the conference organizers, especially the International Advisory Committee, the International Scientific Programme Committee, the Organizational Committee and the Local Executive Committee for their commitment shown in organizing and holding the conference. The importance of initiatives such as FR13 for sharing information and discussing relevant issues concerning the development and deployment of innovative fast reactors and related fuel cycles was emphasized in all four opening speeches.

Several considerations on the role of nuclear power in the actual context of increasing energy consumption and concerns for its impact on the environment were shared in the opening lectures. A common element which can be highlighted from the four opening speeches is the general belief that nuclear fission will continue to play an important role to meet future energy needs, complying with the expectation for low impacts of energy use on climate, environment and health. In this context, the future development of innovative nuclear energy systems and closed fuel cycles is regarded as a necessary step to ensure a long term sustainable energy supply. However, besides sustainability, technology development will have to create the basis to guarantee the respect of the modern requirements of safety, economic competitiveness, proliferation resistance and physical protection.

As underlined by the speakers, the development of fast neutron reactors and related closed fuel cycles lead to impressive advantages, as it was already pointed out by Enrico Fermi. The breeding and the recycle of the spent fuel allow full utilization of the nuclear fuel and significantly improved waste and back end management. Nowadays, although Pu breeding is still considered in many countries a driver for the development of fast reactors, new versatile innovative systems are being developed to implement different fuel cycle strategies. To reach these objectives, large R&D programmes are carried out worldwide, and significant work has to be performed to achieve complete industrial maturity. In respect of the historical development of fast reactors technology, it is worth mentioning the interesting review given by Mr. Michel of the evolution of the key drivers occurring during the past decades, from the very pioneering age until the present day, including also a summary of the major events and technical advances made since the previous conference (FR09).

As a representative from the host country, Mr. Bigot shared with the participants some keynote aspects on the fast reactor programme in France, which is currently one of the largest in terms of objectives and dedicated resources. More specifically, the roadmap foresees the construction of the GEN-IV industrial prototype, ASTRID, which is heavily based on extensive international cooperation.

The opening speeches also addressed issues concerning the relevant impact that the 2011 accident that occurred at the Fukushima Daiichi nuclear power plant had on the public vision and acceptance of nuclear power. However, as also remarked by the IAEA Director General Mr. Y. Amano, despite some predictions to the contrary, the global use of nuclear power is expected to continue to grow in the next few decades. In this respect, Mr. Amano stressed the importance of regaining public confidence in nuclear energy, which is essential for creating the basis for future expansion of nuclear energy utilization.

The necessity for international collaboration was strongly underlined, especially by Mr. Amano and Mr. Sagayama, Former Chairman of GIF. Mr. Amano remarked that historically, the IAEA has been, and remains, a unique collaboration forum, including all the countries in the world engaged in fast reactor development programmes, for ensuring continued progress in fast reactor technology, providing an umbrella for knowledge preservation, information exchange and collaborative R&D. The first decade of GIF was reviewed by Mr. Sagayama, who summarized the history of the initiative, reviewed the four generations of nuclear energy systems, the objectives of GEN-IV systems development, the choice of the six GEN-IV systems and the current activities and, last but not least, the establishment of safety design criteria (SDC) for new fast reactor designs after the Fukushima Daiichi accident.

NATIONAL AND INTERNATIONAL FAST REACTOR PROGRAMMES

National and international fast reactor programmes currently undertaken worldwide were presented during the first day plenary session of the conference.

China

In China, the national programme on fast reactors is part of an ambitious objective to reach, by 2050, a nuclear electricity ratio of 25% out of a total capacity of 2500 GW(e). The main goals of this strategy are to meet the need for growing energy consumption and to improve the structure of the energy sector in the country. The first step of this programme is the acquisition of the necessary experience through the operation of the China Experimental Fast Reactor, which was connected to the electric grid in 2011. The roadmap foresees the construction of the demonstration reactor CFR-600, currently under development, which will act as bridge to the commercial fast reactor CFR-1000, planned to be developed by 2030.

France

France is actively involved in fast reactor projects, which are being developed to address problems related to growth of plutonium inventory, natural uranium utilization and transmutation and burning of minor actinides. Drawing on extensive experience in design, construction and operation of SFRs, the last reactor under operation Phénix was definitively shut down in 2009. Nowadays, based on this significant experience, the Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) is currently in the phase of conceptual design. The reactor is designed as a pool type sodium cooled reactor with a power output of 600 MW(e). The design takes into account the lessons learned from the Fukushima Daiichi accident, and extensive international R&D collaboration is established in support of the project. As a long term fast reactor option, France is also considering gas cooled fast reactor (GFR) technology. In particular, the CEA is involved in the ALLEGRO consortium, which is aimed at the design and construction of a GEN-IV GFR demonstration plant.

India

In India, the successful experience gained with the FBTR, in operation since 1986, represents the main technology basis for the fast reactor programme in the country, which is currently focused on the construction of the 500 MW(e) Prototype Fast Breeder Reactor (PFBR). The PFBR works are at an advanced phase, sodium filling of the primary system is foreseen as being completed by the end of 2013, and the reactor is expected to reach criticality by 2014. The PFBR project will provide the experience for the construction of several commercial fast breeder reactor units, which will benefit from the construction and operational experience and save in capital cost, reduce construction time and enhance safety. The final goal of the Indian fast reactor programme is the deployment of metal fuel breeder reactors, which would allow reducing the doubling time from 30 years for mixed oxide fuel to 12 years (8 years for improved metallic fuel without Zr). This programme foresees fuel pin and subassembly irradiation in the FBTR, the realization of a 120 MW(e) experimental fast reactor followed by the construction of a 1000 MW(e) metallic fuel reactor.

Japan

After the Fukushima accident, nuclear power represents a central topic for public discussion in Japan, and a new nuclear energy strategy is still in the process of deliberation. At present, the Joyo experimental fast reactor and the Monju prototype reactor are shut down. However, the country is in the process of reviewing its strategy for SFR R&D with the aim of making it compatible with the new safety regulations as well as the new energy strategy. The GEN-IV loop type Japan Sodium Fast Reactor (JSFR) has been under development within the FaCT project and is, at present, suspended pending new governmental deliberations. The JSFR design has been recently reviewed after the Fukushima accident, in order to implement the main lessons learned.

Russian Federation

Currently, nuclear power plants in the Russian Federation cover about 17% of domestic electricity needs, with a total capacity of around 24 GW(e). The Energy Strategy of the Russian Federation forecasts nuclear power capacity to increase to 52–62 GW(e) by 2030, and to 100–140 GW(e) by 2050 to ensure the sustainability of energy supply for the Russian economy. Analysis of the different requirements to implement future strategies results in the necessity of deploying fast reactors and closed nuclear fuel cycles, an area in which the Russian Federation has extensive experience and know-how. As for SFRs, an aggregate experience of 140 reactor-years has been gathered through the operation of the BR-5/10, BN-350 and BN-600 reactors. On the basis of this extensive experience, the BN-800 is now under construction, and the commercial BN-1200 reactor is in an advanced stage of design. The Russian Federation is also strongly involved in heavy liquid metal cooled reactors projects, specifically the BREST-300 and the SBVR-100 concepts, which should enter into operation by the end of this decade.

United States of America

Key current issues on nuclear power in the USA are the evaluation of the Fukushima Daiichi accident, the strategy for the management and disposal of used nuclear fuel and high level radioactive waste and the development of small modular reactors. The research programme to support advanced reactors and fuel cycle options in the USA mainly focuses on long term, science based R&D activities that support increasing the performance of fast reactor technology. Major objectives include the development of technologies to improve reliability, safety and plant longevity. Advanced materials, inspection technologies, advanced energy conversion systems, compact reactor concepts, fuel handling systems and advanced modelling and simulation code development are examples of aspects being investigated in national laboratories, universities and industry. All of these activities are carried out in the framework of wide international collaboration.

European Union

In Europe, the Sustainable Nuclear Energy Technology Platform has been established to promote research, development and demonstration of the nuclear fission technologies necessary to achieve the goals of the European Strategic Energy Technology Plan (SET-Plan). Within the Sustainable Nuclear Energy Technology Platform, the European Sustainable Nuclear Industrial Initiative is aimed at supporting the development of innovative fast reactor systems that meet the long term goals of the SET-Plan. The SFR is considered as a reference technology; the roadmap for its development is embodied in the ASTRID project. Lead cooled and gas cooled fast reactors represent the alternative technologies, for which conceptual design of demonstration plants — ALFRED and ALLEGRO, respectively — are currently under way.

NEA

The NEA carries out a number of activities to support the development of fast neutron reactor technology and related fuel cycles. In particular, recent initiatives are focused on SFR studies, heavy liquid metal technologies, innovative fuels and materials, advanced fuel cycle scenarios, partitioning and transmutation, and recycling technologies. Strategic issues such as experimental facilities for fast reactor safety analysis, new trends in fuel cycle and fuel cycle transition scenario studies are the subject of investigation. Last, but not least, modelling and validation activities are performed, with particular attention to the needs of nuclear data for fast reactors and databases of integral experiments.

Panel events

Two panel events devoted to safety design criteria for fast reactors and sustainability of advanced fuel cycles were held during the conference.

The first panel on safety, representing a diverse set of Member States with fast reactor programmes, presented and discussed various considerations related to SDC. Most of the discussion centered on SDC for SFRs, but more generally, the discussion could be applied to all fast reactor concepts.

The details of a recent effort conducted by the multilateral Generation IV programme to develop SDC for the SFR were discussed. The SFR SDC were derived from the Generation IV programme goals and were developed in a manner that was consistent with the hierarchy found in the IAEA safety standards. Important to this effort was the fact that the SDC took into account the unique aspects of the SFR. This resulted in a number of current light water reactor (LWR) based criteria being modified and additional criteria specific to SFRs were added. The SDC were viewed as a good first step, but needed vetting with other stakeholder organizations and required more specific definition.

The panellists shared their national views on the need for SDC, especially for Generation IV reactors. A common theme was that there needed to be a clear differentiation between the criteria for Generation III reactors and Generation IV reactors. Otherwise, it would be difficult to show how safety is being improved in the Generation IV systems.

Other common themes emerging from the discussion included:

- The SDC are very important for both the designer and regulator and a dialogue between the two is necessary to come to a common understanding.
- While the safety principles should be common, the technology specific SDC should consider the unique aspects of fast reactors, including coolant, fuel, fuel assembly geometry and other unique aspects.
- The SDC need to address both accident prevention and consequence mitigation.
- The SDC need to include ‘Fukushima lessons learned’, especially criteria related to complete loss of electrical power, loss of heat sink and the design basis for external events.

With respect to accident prevention, the panellists noted the importance of passive safety concepts for both decay heat removal and protection from reactivity insertion accidents. For consequence mitigation, many of the panellists discussed the need to maintain long term coolability and the use of core catchers to ensure coolability.

During the discussion with the audience, important safety concepts were discussed. Of particular note was the importance of not only having quantitative SDC, but also have a robust demonstration that the fast reactor satisfies the criteria. It was noted that it would be easier to convince the stakeholders of the safety case if the argument were simple.

The second of the two panels was held on 5 March. There were eight panellists, six from Member States and two from international organizations.

Sustainability was discussed in terms of the social, environmental and economic perspectives, which arise from the original Brundtland definition of sustainability. The panellists presented their perspectives of the need to move towards a sustainable future, involving better use of uranium; reduction in high level radioactive waste; and the safe, secure and economic operation of nuclear reactors and the fuel cycle. In all cases, it was considered that sustainability in the long term must involve fast reactors and a closed nuclear fuel cycle, although both the Republic of Korea and the IAEA pointed out that these are clearly national decisions and there will not be a single solution for all countries.

In fact, it was emphasized that, given the size of the nuclear programme in various countries, only the larger countries are likely to implement a full version of the closed fuel cycle. This raises the issue of how smaller nuclear countries will gain access to the reprocessing and recycling capabilities which are needed to deal with their spent fuel. Given this situation, there was strong support for continued international collaboration, such as that provided by the GIF and INPRO.

The progress made towards achieving a closed nuclear fuel cycle also requires advances in technologies, such as partitioning and transmutation, fuel design, new waste matrices for encapsulation of high level waste and developments in reprocessing technologies. Different types of fuel cycle involving fast reactors were also discussed, including a fully closed cycle using Pu fuel and fast reactors and other symbiotic fuel cycles involving both fast reactors and LWRs. It was recognized that the transition from the LWR cycle to a fast reactor cycle will be difficult. The European Commission pointed out that there were a number of risk factors that must be addressed on the way to sustainability. Several speakers indicated the importance of public acceptance for any such transition and therefore the need for discussion with the public in an open manner on the advantages and disadvantages of fast reactors. It was also noted that, in any such discussions, the impression should not be given that there were problems with the existing LWR cycles.

While the panel members outlined national approaches and the development of technologies within their countries, the drivers for the move to introduce fast reactors and recycling seem to vary between countries. In some cases, that need is expressed mainly in terms of resource utilization and ensuring a long term supply of fuel, as well as a means to deal with plutonium inventories. In other cases, the driver seems to be the need to manage radioactive waste, to reduce the amount of material requiring removal to deep geological repositories and the demonstration to the public that the industry is not leaving an undue burden on future generations.

Overall, it was agreed that there are clear advantages in introducing fast reactors and a full recycle of Pu for the long term sustainability of nuclear energy. However, the time frames for such a move are uncertain and the way such a process will be introduced by many countries is still not decided. In the interim, continued research is needed on the technologies that will support a full recycling process and international cooperation remains a primary means by which progress can be achieved.

FAST REACTOR DESIGNS: GOALS AND PATHS OF PROGRESS

A broad range of fast reactor progress paths, reactor types and candidate technologies are currently being assessed worldwide within the different fast reactor development programmes. These efforts were presented during the five technical sessions devoted to the technical track Fast Reactor Designs: Goals and Paths of Progress, which included 27 papers presented orally and one poster session presenting 21 contributions.

The technical sessions focused on the different design approaches and strategies characterizing the different fast reactor projects, which generally result from performance, safety, economics and fuel cycle considerations. The main emphasis was put on liquid metal cooled reactor concepts, i.e. SFRs and heavy liquid metal cooled fast reactors, but also advances in GFRs and molten salt reactors were presented.

Major trends in advanced SFR designs were highlighted during the technical sessions. As for innovative core designs, ‘low void’ cores are developed for instance within the framework of the ASTRID project in France and for the BN-1200 in the Russian Federation, whereas conventional ‘positive void’ cores are foreseen for the JSFR in Japan and the Indian commercial fast breeder reactor designs. In general, common approaches can be identified in the areas of passive and diversified shutdown systems, decay heat removal systems and air ultimate heat sink. These last aspects became even more important after the Fukushima accident and the lessons learned currently represent an important drive force for new R&D and design innovations. Different fuel forms are being assessed, from the more conventional mixed oxide (MOX) form to nitride and metal fuels.

In France, drawing on its long experience with SFR technology, high levels of requirements have been set for the 600 MW(e) pool type demonstrator reactor ASTRID, for which wide international cooperation has been established to develop innovative design solutions which allow achieving the objectives of improved performance, enhanced safety and higher standards in terms of operability and transmutation capabilities. It was remarked that meeting these high requirements set for ASTRID and serving R&D needs will require increased industrial and international collaboration. New promising innovation options have been identified during the first phase of the ASTRID conceptual design (2010–2012); evaluations on low sodium void effect core demonstrated the safety potential and therefore this concept is retained for the conceptual design, which will be delivered by the end of 2015. The pre-conceptual design also identified innovative solutions for the nuclear island architecture, the primary and secondary circuits, steam generators and sodium/gas heat exchangers. These innovations will be further developed in the next phases of the design studies, keeping a strong interaction between design and R&D.

Contributions from India focused on oxide fuel and metal fuel fast reactors, both under development in the country as succeeding phases of the fast reactor programme. With regard to the first, based on the PFBR, which is currently in an advanced phase of construction at Kalpakkam, India will build in the next decades two commercial fast breeder reactor units. Several aspects were presented of the structural mechanics and thermohydraulic designs, of particular importance for guaranteeing the targeted plant performance and the required high safety levels. As for metal fuel fast reactors, planned to be deployed in India in the ultimate phase of the fast reactor programme, physics design and safety studies of a 320 MW(th) experimental metal fast breeder reactor were presented.

In Japan, the loop type JSFR design was reviewed in the light of the Fukushima accident, in particular the loss of energy supply and loss of heat sink. Safety design requirements of the JSFR, which are set to achieve a safety and reliability level equal to advanced LWRs, include innovative technical solutions such as passive shutdown systems and natural convection decay heat removal systems. For external hazards, evaluations have been conducted and have shown that the JSFR already had sufficient safety design features to counter major earthquakes and the long term loss of all AC power. International cooperation will be of particular importance in developing safety design criteria and measures.

The SFR programme in the Republic of Korea is aligned with GEN-IV goals and it currently foresees the conceptual design of a 150 MW(e) prototype SFR, the erection of an R&D infrastructure, including large sodium loops, and fabrication and irradiation of U-Zr based metal fuels. The Korean contribution to this track highlighted the main innovative SFR design features, which include a superheated steam Rankine power conversion system, double-wall steam generators, and the coexistence of active and passive heat removal systems. Solutions for under-sodium inspection techniques based on ultrasonic waveguide sensors are also foreseen. Contributions also included the presentation of other fast reactor concepts being assessed, e.g. the ultra-long cycle fast reactor and the multipurpose experimental sodium cooled fast reactor, which could be a first step before the construction of the SFR demonstration plant.

The Russian Federation is currently involved, under the Federal Target Programme on New Generation Nuclear Power Technologies for the Period 2010–2015 and for the Long Term up to 2020, in the development of the commercial size BN-1200 SFR. The design is based on the experience gained with the BR-5/10 and BOR-60, the operation of the BN-350 and BN-600 and the construction of BN-800 reactors. For the BN-1200, specific design criteria are considered, which focus for instance on the natural behaviour in the case of unprotected loss of force circulation. In this respect, the quantitative objective — sodium non-boiling — is not achieved if oxide fuel is considered, whereas it is achieved with a nitride fuelled core. However, at the current development stage, the fuel form has not yet been decided.

In the USA, where the main efforts are focused on long term, science based R&D activities that support increasing the performance of fast reactor technology, a small size (100 MW(e)) advanced SFR (AFR-100) is being developed. The concept is characterized by a more compact design, higher performance and longer fuel cycle (>20 years). The AFR-100 is designed to be transportable from pre-licensed factories and installed in small grids and it does not need frequent on-site refuelling. In order to develop this concept, various innovations to improve safety are introduced in the design.

Roadmaps for the development of lead cooled fast reactor (LFR) concepts were also presented, which include the design of the first prototype reactors, i.e. the accelerator driven system prototype Multi-purpose Hybrid Research Reactor for High-tech Applications (MYRRHA) in Belgium, the China Lead Alloy Cooled Research Reactor (CLEAR-I) in China, the demonstrator ALFRED in Europe and the SVBR-100 and BREST-OD-300 in the Russian Federation. The contributions that focused on heavy liquid metal cooled systems highlighted the great potential of these systems, and from the other side the need for R&D activities to increase technical maturity and cover the technological gaps.

In Belgium, the fast reactor development programme focuses on MYRRHA, a flexible fast spectrum research reactor (50–100 MW(th)), conceived as an accelerator driven system and able to operate in both subcritical and critical modes. It features a proton accelerator of 600 MeV, a spallation target and a multiplying core with MOX fuel, cooled by liquid lead–bismuth. This system will be used for fuel developments, material developments, doped silicon, radioisotope production and fundamental science applications. Next to these applications, MYRRHA will demonstrate the accelerator driven system full concept by coupling a high power proton accelerator, as well as being able to significantly contribute to the development of LFR technology.

Contributions from China in this area focused on the recently initiated activities on lead based fast reactor systems. In particular, the Chinese Academy of Sciences has been involved in the development of an accelerator driven system and lead alloy cooled fast reactors since 2011. One of the main goals is to build the first reactor, CLEAR-I, before 2020, which is featured as a 5–10 MW accelerator driven system research facility, including a 50–250 MeV accelerator coupled with a liquid lead–bismuth cooled core. Conceptual designs of LFR have been performed and lead–bismuth experimental loops have been constructed for material tests and thermohydraulics and safety tests.

The European project LEADER, led by Italy, is aimed at the demonstration of industrial scale pure lead cooled fast reactor technology, combining the inherent favourable characteristics of the coolant with innovative design solutions. One of the aims of the LEADER project is the conceptual design of the prototype ALFRED, a 300 MW(th) pool type system. The comprehensive approach adopted for the core design investigated within the LEADER project proved to effectively drive the design to the fulfilment of the aimed sustainability performances and the respect of the design constraints for the robust implementation of the inherent safety principle.

In the Russian Federation, the SVBR-100, a 100 MW(e) lead–bismuth cooled fast reactor, is at an advanced stage of design. The main results of several evaluations on core physics and thermohydraulic characteristics of this concept were described. Several analyses were also recently conducted to assess the applicability of thorium based fuels in SBVR-100 units. The Russian Federation is also involved in the design of a pure lead cooled reactor, the BREST-OD-300, a nitride fuel fast reactor operating in a closed uranium–plutonium cycle. The BREST-OD-300 reactor is a pilot demonstration fast reactor of 700 MW(th) and 300 MW(e), with two heat removal circuits which use a water–steam mixture of subcritical parameters as the secondary circuit fluid. The design is based on the experience of nuclear power evolution and the research conducted in the country in this area over decades.

GFR and molten salt fast reactor concepts are also being studied in a number of countries. Development paths were presented along with the status and main characteristics of the different reactor concepts.

FAST REACTOR TECHNOLOGIES, COMPONENTS AND INSTRUMENTATION

Contributions to the technical track "Fast reactor technologies, components and instrumentation" were presented in five different sessions, which overall included 26 papers presented orally and a poster session which included 22 contributions.

Contributions to the technical sessions mostly presented technology advances arising from innovative fast reactor development programmes and related R&D activities. The main goal of these efforts is to find innovative engineering solutions which allow achieving the different design objectives in terms of performance, operability and maintenance, reduced capital cost and enhanced safety levels. With regard to the latter, measures against severe accidents and external hazards are especially sought in the light of the Fukushima accident. The strong link between all these aspects of the design of innovative fast reactors was remarked in the numerous contributions related to this technical track.

Significant advances in the area of SFR technology are being achieved in China through the China Experimental Fast Reactor project, which represents the technology base for the development and deployment of the commercial SFR fleet planned in the country over the next decades. An example of recent innovation presented at the conference is a mobile detector system used for burnup measurement of the spent fuel assembly and for on-site applications. Another contribution from China concerned the status of the hydrogen detection technique for the China Experimental Fast Reactor steam generator protection system, including the main aspects of its physical background and engineering implementation.

Large scale international R&D collaboration is set up to support the ASTRID project in France. Among the several innovative technical solutions being assessed in the frame of its conceptual design is the power conversion system, which is the focus of a number of studies aimed at reducing the footprint of the plant as well as the risk of energetic sodium–water reactions. The reference option is currently a Rankine saturated steam cycle and feedwater reheating, for which the design of the steam generators is of particular importance in order to reduce the sodium–water reaction occurrence as well as to limit its consequences in the case of a hypothetically violent reaction. A Brayton cycle with pure nitrogen at a pressure of 180 bar is also being studied as an alternative option; preliminary studies proved its high potential in terms of performance, safety features and economics. On the basis of past experience, large electromagnetic pumps are being developed in collaboration with Toshiba, with the aim of exploiting several advantages in terms of operation and maintenance. Other R&D activities are performed in France to improve in-service and inspection methods, develop core catcher technology and improve understanding of key safety related phenomena, such as the behaviour of sodium fire aerosols during atmospheric dispersion.

Contributions from India summarized the status of development of the SFR's main components, based on feedback from the PFBR experience on key systems such as the grid plate, roof slab, thermal insulation panels, safety vessel, fuel transfer machine and primary pipes. The importance of the development of reliable methods for the integration of components fabricated by different industries was remarked on. With regard to the PFBR construction, a number of innovations in the equipment design, fabrication and erection were presented, with particular attention to the high standard of quality control and quality assurance during design, material procurement, forming, welding, fabrication, handling and finally testing. A specific example was given with a contribution that focused on the design and fabrication of serpentine tube type sodium–air heat exchangers for the PFBR decay heat removal system.

In Japan, even though the JSFR design already included measures against severe accidents and passive safety systems, e.g. passive shutdown systems and natural convection decay heat removal systems, several evaluations were conducted on severe external events after the Fukushima accident. The analysis shows that the inclusion of seismic isolators allows the plant components to have a sufficient seismic margin against Fukushima-like events. It was also remarked that the JSFR can maintain long term decay heat removal capability thanks to full natural convection decay heat removal systems. Further improvements, for instance the implementation of measures against tsunami and on alternative decay heat removal systems, are also being evaluated. As for innovative reactor component development, a once-through sodium heated steam generator with double-wall straight tubes is designed for the JSFR, for which dynamic tests are planned to be carried out in the Advanced Technology Experiment Sodium Facility at the Japan Atomic Energy Agency. Relevant technology advances are also expected to be achieved through the restart and operation of the loop type Monju SFR. For example, a highly sensitive failed fuel detection and location technique has been developed in the framework of the core R&D programme for commercialization of the fast breeder reactor by utilizing Monju. Innovative solutions in the area of safeguards for the Monju reactor were also presented.

The evolution of the engineering solutions from the previous BN-series designs to the BN-1200 was presented by a representative from the Russian Federation, with particular attention to the reactor design, fuel performance, steam generators and refuelling system. The integral BN reactor concept was firstly developed within the BN-600 nuclear power plant, and maintained without major changes for the BN-800 and BN-1200 designs. The main improvements in the BN-800 design concerned the seismic safety enhancement and the inclusion of a core catcher in the reactor vessel. As for the BN-1200, further innovations consisted of a primary circuit completely integrated into the reactor tank and an in-reactor shielding made of boron carbide located around the core. The experimental validation of the mechanical equipment, in particular of the refuelling and control rod drive mechanisms, is also an important part of the development activities for the BN-1200. The main objective is to guarantee economic competitiveness (in comparison to a PWR reactor of the same power) by reducing overall dimensions and construction materials.

The Republic of Korea is currently engaged in relevant R&D programmes on SFR technology, in particular in support of the current design of the 150 MW(e) GEN-IV prototype reactor. A relevant part of this programme is the improvement of in-service inspection and repair techniques; in this area, a new concept for a plate type ultrasonic waveguide sensor and inspection technique for under-sodium viewing has recently been developed.

Research efforts in the USA are focused on technology options which allow significant performance improvements and reduction of the capital cost of advanced reactors; in order to reach this goal, innovative solutions are sought in the areas of concept development studies, advanced structural materials and components, non-conventional power conversion systems and advanced modelling and simulation. Other important research activities are carried out to improve safety as well as to develop high performance fuels and key technologies for fast reactors, e.g. under-sodium viewing and oxygen control. Another contribution from the USA is provided in the frame of the SFR component design and balance-of-plant project of GIF.

With regards to LFR technology development, contributions to this track highlighted progress resulting from the European programme LEADER. More specifically, the main design options and the different solutions adopted for the demonstrator plant (ALFRED) and the European Lead Fast Reactor (ELFR) were presented. The design of the steam generator units is key for LFRs, especially for preventing water–liquid metal interactions and managing the consequences of any possible leaks. A double-wall bayonet tube steam generator is under development for the ALFRED reactor, while an innovative once-through flat spiral tube concept is foreseen for the ELFR. Innovations for the reactor configuration, core design and decay heat removal systems were also presented.

FAST REACTOR SAFETY: POST-FUKUSHIMA LESSONS AND GOALS FOR NEXT GENERATION REACTORS

The accident that occurred in March 2011 at the Fukushima Daiichi nuclear power plant drew renewed attention to the safety of current and future nuclear energy systems. In the light of this event, safety features of fast reactors was one of the most discussed topics of the FR13 conference. A specific track, Fast Reactor Safety: Post-Fukushima Lessons and Goals for Next Generation Reactors, which included five technical sessions, was devoted to this topic. Fifty contributions were received in the technical sessions, of which 27 were presented orally and 23 presented as posters. SFRs and LFRs were mainly considered, with a few presentations on GFRs.

Safety analyses of the China Experimental Fast Reactor were presented, with special focus on transient and accident scenarios such as unprotected transient overpower accidents and unprotected loss of flow, both investigated by means of the SAS4A simulation code.

In France, safety orientations for the ASTRID conceptual design are mostly based on past experience gained in SFR technology, the main lessons learned from the Fukushima accident and intense R&D supporting programmes. More specifically, major safety objectives, the implementation of safety design methods, severe accident prevention and mitigation measures were summarized, emphasizing the basically deterministic approach adopted in the safety assessments. A wide range of analytical activities including modelling, code development and verification, validation and qualification, with the support of large experimental programmes aimed at gathering new data in support of severe accident studies, have also been presented by French representatives. Some of these activities are of particular relevance for international cooperation, especially for benchmark analysis and sharing of experimental infrastructures. The French Nuclear Safety Authority has also been undertaking actions to identify benefits and drawbacks for the six GEN-IV systems and to analyse the safety orientations proposed for ASTRID.

The increased attention to the enhancement of safety after the Fukushima accident was also discussed by experts from India, who presented the innovative measures foreseen for the safety upgrade of the PFBR, especially in terms of severe accident management strategies. The PFBR already faced a tsunami in December 2004 while under construction and, as a consequence, changes were included in the design, for instance to accommodate the water level rise during a tsunami. The Fukushima accident provided additional feedback on design features needed to face severe natural events. To support this process of safety improvement, a large number of experimental and numerical activities are carried out to improve understanding of safety related phenomena and to develop new technologies for the prevention and mitigation of severe accidents. The importance of the harmonization of safety standards within international initiatives was also noted by Indian representatives.

In Japan, after the events occurred at the Fukushima Daiichi nuclear power plant, safety is clearly the major aspect which determines current and future strategies for nuclear power development; in the light of this accident, important considerations have been given to the safety of innovative SFRs. As a result, the main lessons learned from the accident are being included in the safety design approach adopted for the JSFR, in terms of basic approach, design extension conditions and practical elimination of certain accidental situations. Design measures foreseen in the JSFR design were presented, especially those conceived against failure of the shutdown system, abnormal reactivity insertion and failure of normal heat removal from the reactor core. The design approach for the reactor containment was also discussed, considering its importance for situations characterized by mechanical energy release as a consequence of re-criticality, deflagration/detonation of accumulated hydrogen induced by fuel debris–concrete interaction or sodium–concrete reaction, large scale sodium leak and fire. Experiments on natural circulation for decay heat removal and measures for guaranteeing in-vessel retention of core disruptive accidents are also an important part of the JSFR safety approach.

Modelling of transients and hypothetical accident scenarios for SFRs, as well as validation of the used codes, is being performed in Germany as a result of the recognition of the importance of validated and qualified simulation tools for the design and licensing of innovative fast reactors to be deployed for transmutation of nuclear waste. An example of these activities is, for instance, the understanding of transient fuel behaviour and post-failure phenomena occurring during and after power excursions, for which several experimental activities were performed in the past years.

Modern approaches for safety assurance of new generation SFRs are being assessed in the Russian Federation, in particular within the framework of the commercial size BN-1200 design, which includes, as a main requirement, the non-evacuation of residents under all possible realistic accident conditions. Safety design of the BN-1200 is based on past experience on the design and licensing of BN series plants; during the 30-year operation of the BN-600 reactor, the regulatory body performed safety reviews leading to improvements in technical operation, safety and economy. The long period of accident free operation demonstrates a level of mastery of this reactor technology and performance of the appropriate supervision from the regulatory body. Specific safety aspects presented during the technical sessions included analytical and experimental studies for the validation of innovative solutions to confine BN reactor melted fuel. Even though the BN-800 and BN-1200 reactors are such that fuel melting has a low probability of occurrence, a core catcher is foreseen in order to prevent the reactor vessel and the guard vessel from melting in the case of such a scenario.

SFR safety and licensing research programmes are also ongoing in the USA, including the identification of safety related gaps and the analysis of existing uncertainties regarding the Department of Energy's capabilities to support an SFR licence application. A wide range of R&D activities is performed in support of innovative fast reactor safety; in particular, materials performance research represents a very important part of this effort, considering its relevance for the licensing process. Such studies demonstrated that advanced materials such as oxide dispersed strengthened (ODS) alloys for cladding, Gr91 and 92 F/M steels, and austenitic alloys such as NF709 for structures can improve the economy, safety and flexibility of SFRs. Safety aspects related to the utilization of thorium fuel in SFRs have also being analysed by the US nuclear industry.

Identification of safety design approaches for lead based alloy cooled fast reactors is pursued in many countries with development programmes in this area. Safety features of LFR systems rely on the favourable characteristics of the coolant, especially the absence of exothermic chemical reactions with water and air, as well as the exclusion of global coolant boiling due to the high boiling temperature of lead and lead alloys.

Preliminary safety analyses of the CLEAR-I — a lead–bismuth cooled system which can be operated in both critical and subcritical modes — have been recently conducted. For all the investigated transient accidents, a large margin to fuel melt was pointed out, owing to the relatively low power density, establishment of the coolant natural circulation and passive decay heat removal system. Efforts are currently devoted to the preparation of licensing works, which will need further activities on lead–bismuth eutectic experimental data and code validation, employment of different computer codes, feedback from past licensing experiences, etc.

Within European programmes on LFR and accelerator driven system technology, Germany recently conducted safety analyses on the ELFR and MYRRHA, especially with the aim of assessing evolution of unprotected loss of flow, unprotected loss of heat sink, flow blockage and reactivity insertion events. In particular, if steam generator tube rupture is being considered, this will require a dedicated scaled facility to experimentally analyse in depth the steam generator tube rupture phenomena.

Safety analyses of representative design extension conditions for the ALFRED reactor were recently performed in Italy, which is the leading country of the LEADER project. Results show very good intrinsic safety features of the design thanks to the natural convection in the primary circuit, large thermal inertia, and prevalent negative reactivity feedbacks to reduce power excursions. In all the analysed unprotected transients, no risk of significant core damage and of transient evolution towards severe accidents was found. Owing to the intrinsic characteristic and the design solutions of the system, enough grace time is left to the operator to take opportune corrective actions for a safe reactor shutdown.

Contributions from the Russian Federation highlighted the main safety features of both the pure lead cooled BREST-OD-300 reactor and the lead–bismuth cooled small size SVBR-100. As for the first, safety analyses on positive reactivity insertion and loss of forced heat removal scenarios have been carried out. These studies have shown that none of these situations leads to fuel melting and coolant boiling. Inherent self-protection is an important objective of the design of the small size modular reactor SVBR-100, which is achieved thanks to the favourable properties of, and the low energy stored in, the lead–bismuth coolant. Further, the design foresees negative reactivity feedbacks, which, combined with a reliable control and protection system, eliminates the risk of prompt neutron runaway. Another inherent safety feature is represented by the safeguard casing of the vessel, which eliminates the risk of loss of coolant accidents and high pressure radioactive releases.

Safety aspects of non-liquid metal cooled fast reactors were also discussed in a technical session included in this track, to complement design issues presented in track 1. More specifically, a contribution from Slovakia concerned recent analytical studies on the coupled reactivity effects of a movable reflector and safety control rods in GFRs.

International initiatives in the area of innovative fast reactor safety were also presented, e.g. the European project SARGEN_IV: Proposal for Harmonized European Practices for the Safety Assessment of Innovative Fast Neutron Spectrum Reactors Considered in Europe, which was launched by the European Commission in November 2010. The SARGEN_IV represents an opportunity to prepare the safety assessment of future innovative reactors. To reach this goal, the project brings together European designers, TSOs, research organizations and utilities involved in innovative reactor development programmes. The next phase of the project will include the test application of the proposed harmonized European practices in order to identify R&D requirements and provide feedback to the harmonized European practices in support of the EURATOM contribution to the preparation of a ‘white paper’ on nuclear safety for European GEN-IV concepts.

FAST REACTOR MATERIALS: ACHIEVEMENTS AND NEW CHALLENGES

The development of high performance materials represents a crucial aspect of fast reactor R&D as they directly affect the performance, safety, reliability and economic competitiveness of fast reactors. The relevant technical challenges in this area, arising from the high temperatures, high neutron doses and aggressive environments that characterize fast reactors, require large R&D programmes which are currently carried out at national level and in the framework of international cooperation. Recent results and achievements in this area were discussed during the four technical sessions devoted to the track Fast Reactor Materials: Achievements and New Challenges, which included 20 oral presentations and 13 poster contributions. The papers presented recent results of irradiation experiments on innovative cladding and wrapper materials, neutron absorbers and core structural and shielding materials, as well as relevant aspects of fabrication technologies, welding and hard facing techniques, surface treatments and corrosion issues.

Material options for SFR steam generators are being evaluated in China, with particular attention to degradation phenomena and service life. Reference materials for such application are T22, T23 and T91 steels, which have been widely used in the past in fossil fuel power plants. Thinning of tube wall thickness due to sodium corrosion and steam oxidation is a particularly investigated phenomenon, owing to its relevance for the steam generator tube life. Degradation of mechanical properties due to thermal ageing, e.g. the reduction of tensile property and reduction of creep rupture, is also an important part of the investigations. These studies revealed that T91 is currently the most suitable material for the long operation (50–60 years) of steam generator units.

The overall picture of the R&D programme on SFR materials was presented by French representatives through a large number of contributions received on this technical track. SFR reference materials are currently austenitic steels for fuel cladding, ferritic steels for wrapper tubes and boron carbide B4C as a neutron absorber. Advanced material options under evaluation include ODS steels and vanadium alloys (V-4Cr-4Ti) for fuel cladding and SiC/SiC material for wrapper tubes. As for the cladding reference material foreseen for the ASTRID project, i.e. AIM1 improved austenitic steel, the main aspects under evaluation are swelling phenomena, mechanical properties, plug/cladding welds, chemical compatibility (Na/fuel) and behaviour during transients. Similar investigations are being carried out for wrapper material (EM10), especially in terms of swelling, embrittlement, mechanical properties, base/wrapper weld and behaviour during transients. The SiC/SiC materials are also intensely studied in France, with a particular focus on chemical compatibility and mechanical characteristics. Investigations on corrosion phenomena and development of welding and hard facing technologies are also a significant part of fast reactor materials research in France.

In India, studies on materials for fast breeder reactor core components are under way. In particular, the combined effects of high temperatures (400–700°C for the fuel clad and 400–600°C for wrapper materials) and high neutron fluence on core materials, which lead to dimensional changes and deterioration in the mechanical properties, are being studied. Criteria for material selection of fuel clad (austenitic stainless steels, ferritic-martensitic (F/M) steels, ODS steels and bimetallic materials) were also discussed, including considerations on manufacturing technology. Irradiation experiments to determine the effect of long term low dose irradiation on the FBTR grid plate material are ongoing or planned. Similar investigations are focused on ferro-boron materials for use as in-vessel shields in fast breeder reactors, which are expected to feature favourable characteristics for their use in SFRs.

In Japan, material research is focused on the development of core and structural materials to improve safety, reliability and economic competitiveness of the JSFR. ODS steels are developed for core materials; for these materials the target performance includes a peak neutron dose of about 250 dpa and a mid-wall temperature of 973 K. Several high burnup tests on ODS steels were carried out in the JOYO and BOR-60 reactors; the tests showed improved radiation resistance, good dimensional stability and excellent creep strength of these innovative materials. Peculiar irradiation behaviour, such as microstructure instability and fuel pin rupture, occurred in BOR-60 irradiation tests; the cause was concluded to be the combined effects of two factors: the matrix Cr heterogeneity and the irradiation temperature exceeding the design one. The development of manufacturing technology applied to the full pre-alloy process to improve homogeneity of the 9Cr-ODS cladding has already started. As for structural materials (materials for the steam generator, reactor vessel and internals), 316FR steel (low carbon nitrogen added 316SS) and modified 9Cr-1Mo steel are in an advanced phase of development.

Germany has been carrying out several lines of R&D activity on fast reactor materials. Austenitic steel (15Cr-15Ni-Ti stabilized) is considered as reference material for fuel cladding, but also alternative options as advanced austenitic, ODS, F/M steel, SiC/SiC are considered. Austenitic and F/M steels are foreseen for reactor vessel and primary system materials, and F/M steels and Ni alloys for the steam generators. Contributions from Germany also presented fabrication and characterization of Al-containing ferritic ODS alloys for improved corrosion resistance. Materials for LFRs (ALFRED and ELFR) are also studied in Germany, with a particular focus on material selection, material compatibility (corrosion/erosion) and corrosion protection barrier development.

An overview of materials performance in SFRs was given by a US representative, including historical background, past experience, and current and future activities carried out in the different countries with active fast reactor research programmes. Materials requirements, selection and performance for in-core and out-of-core components in SFRs, for instance advanced materials such ODS alloys for cladding, Gr91 and 92 F/M steels and austenitic alloys such as NF709 for structures, were widely discussed. The contribution also stressed the importance of the validation of materials performance for SFR licensing, the need of a substantial database for all of these materials and the global effort to develop the needed information through experimentation and modelling. A contribution received from the USA presented the development efforts for advanced 9Cr F/M steels and austenitic stainless steels for SFRs.

FAST REACTOR FUELS AND TRANSMUTATION TARGETS: DEVELOPMENT AND IRRADIATION EXPERIMENTS

Track 5 of FR13 conference was devoted to Fast Reactor Fuels and Transmutation Targets: Development and Irradiation Experiments, for which 35 papers were received, 21 papers were presented as oral presentations in four technical sessions and 15 papers were presented in the poster session.

The first session of track 5 started with the overview presentations of national fast reactor fuel development strategies of France, India, Japan, Republic of Korea, USA and the European Commission. Other sessions had presentations related to fuel design involving studies on minor actinide consumption rate by moderators, new fuel concepts for GFRs, fuel clad chemical interaction model in metallic fuel and actinide oxide dispersive metallic fuel. These studies have been carried out to investigate fuel performance from various viewpoints, with goals of attaining high burnup, high minor actinide consumption rates and improved thermal properties. The combination of fast reactor types and new concept fuels is expected to generate improved fuel performance.

The accident at the Fukushima nuclear power plant has impacted the R&D of fast spectrum nuclear fuels and materials in Japan. However, the conceptual design study and development status of fast breeder reactors with MOX fuel in Japan has progressed. The results of the METAPHIX metal fuel irradiation programme were presented. The chemical compositions of metal fuel, U-Pu-MA-RE-Zr before and after irradiation were analysed and their uncertainties discussed.

Fast reactor fuel development in India has progressed with research on multiple fuel types: carbide, oxide, metallic and CERMET. Mixed carbide fuel is used in the FBTR and MOX fuel will be the driver fuel for the PFBR. India is also considering the use of metal fuel in fast breeder reactors. Two design concepts for the metallic fuel development programme are being considered, e.g. a mechanically bonded pin with U-15wt%Pu alloy and a sodium bonded pin with U-15wt%Pu-6wt%Zr alloy. A programme has been initiated for the irradiation testing of sodium bonded U-Zr alloys in the FTBR. Preliminary work has also been initiated on U based CERMET fuels.

There is a long history of fast spectrum nuclear fuel development in the USA and in recent years the goal has been the development of transmutation fast reactor fuel. Metallic based fast reactor fuel technology development is focused on R&D to gain a fundamental understanding of metallic fuels containing minor actinides. Ceramic oxide fast reactor fuels are considered as a backup option. Irradiation of minor actinide bearing metallic (AFC-2 A and B) and oxide (AFC-2 C and D) fuels have been conducted and post-irradiation examination is under way.

The strategy for advanced fast reactor fuel development in the Republic of Korea is moving forward with the development of metallic fuel, U-Pu-MA-Ln-Zr, which uses recovered TRU through pyro-electrochemical processing of spent LWR fuel. To prevent interaction between metal fuel and cladding, Cr electroplating on the inner surface of the cladding is currently under development. Fabricated metal fuel rods, including Cr plated barrier cladding, have been irradiated in the HANARO research reactor and post-irradiation examination is currently ongoing.

There is a long history of development of fast reactor fuels in France. Recently, a critical look at the conceptual designs of fuel subassembly, control and shutdown rods and absorber elements, along with the lessons learned from the operation of French fast reactors (Phénix and Superphénix especially) and the associated irradiation tests, has yielded improved and even innovative fuel and absorber assembly design concepts for the 600 MW(e) ASTRID. Thanks to an ambitious national programme, strong international partnerships and fruitful collaborations within the framework of European projects and the GIF, a significant set of results on minor actinide bearing oxide fuels is now available.

The CEA (France) has also designed fuel elements for GFRs. The novel feature of this design is sandwich cladding and a buffer bond. The fuel ‘meat’ consists of U-Pu-C fuel pellets. The sandwich cladding consists of three layers: an inner SiC/SiC composite layer, a middle metallic layer, and an outer SiC/SiC composite layer. The buffer bond is a high porosity carbon based braid. The SiC/SiC composite cladding is resistant to irradiation but prone to micro-cracking and leaktight fabrication is difficult. Including a metallic liner

(50–100 μm Ta/Nb) within the sandwich improves the ductility and weldability of the cladding. Fabrication has been demonstrated and characterization completed by the CEA. End closure and irradiation programmes are in progress.

Given the proven safety performance of oxide and metal fuels, and the progress made in nitride, carbide, and minor actinide bearing oxide and metal fuel research programmes, improvements in knowledge and understanding of the safety performance of these fuels can be made and are under way. Significant progress has been made in the fabrication, characterization and understanding of the irradiation performance of uranium and plutonium driver fuels. In addition, in support of the fuel cycle closure goals of many national programmes, the inclusion of minor actinide materials in many fuel compositions has been demonstrated by multiple national programmes. Several options in both homogenous and heterogeneous transmutation of minor actinide fuel systems now exist for consideration in fast spectrum reactor designs. Modelling and simulation of fuel properties and underlying mechanisms has made major progress in the last decade. To take maximum advantage, strong coupling between modelling and simulation efforts and experiments is required.

FAST REACTOR FUEL CYCLE: PROCESSES AND DEMONSTRATIONS, INCLUDING PARTITIONING AND TRANSMUTATION

Five technical sessions were devoted to the track on Fast Reactor Fuel Cycle: Processes and Demonstration including Partitioning and Transmutation, which included 36 papers, 25 presented orally, and 11 papers in the form of posters. The contributions mainly dealt with the fast reactor fuel cycle, involving multiple recycling of plutonium and partitioning and transmutation processes which aim at the separation and recycling of minor actinides from the spent fuel or high level liquid waste. Recycle of minor actinides enables not only efficient use of resources but also reduces the waste volume and radiotoxicity. The importance of partitioning and transmutation processes for increasing and sustaining nuclear energy growth in the world has been realized in several countries. Accordingly, developments in advanced partitioning processes, as well as minor actinide bearing fuels, have received enhanced interest in recent years. Thus, many countries with large nuclear installations are exploring advanced separation technologies for the next generation fuel cycle. Both aqueous and pyrochemical processes have been investigated to strengthen proliferation resistance and mitigate the environmental burden by managing transuranic elements. Overall, the presentations demonstrated that separation processes are now available to match all the advanced fuel cycle options involving both heterogeneous and homogeneous transmutation of minor actinides. Several fuel concepts have been tested in dedicated irradiation tests.

In India, several studies on the fabrication of fuels for fast breeder reactors are currently being performed. These activities are aimed at the development of a safe and efficient fuel element fabrication line, which features conveyors used for transfer of fuel pellets, a remotely operated trolley for movement of clad tubes, simultaneous loading of MOX and axial blanket fuel pellets, separation of pellet handling and element handling, special systems for tube handling and ultrasonic and laser decontamination.

Transmutation scenario studies are performed in France, where fuel cycle processes and technologies have already been proven industrially for oxide fuels on the basis of LWR experience. Also, R&D on different techniques for separation of minor actinides and the fabrication of minor actinide bearing fuel materials is under way. These activities are also performed within the framework of international initiatives, such as the FP7 Euratom collaborative project ACSEPT, which deals with spent fuel reprocessing, reprocessing techniques and related basic research. The impact of minor actinide transmutation options on geological disposal is also being evaluated. As for SFR fuel technology, French contributions presented different issues related to fuel fabrication, based on the past experience gained in the country over the past decades and R&D challenges for industrial SFR MOX fuel treatment.

A contribution from Germany focused on the fuel cycle associated with the molten salt fast reactors as highly efficient transmutation systems. First, conceptual studies show that the molten salt fast reactor offers very attractive features for efficient transmutation and significant advantages due to liquid fuel and online refuelling and reprocessing. In contrast, significant developments are required on the way to its application, owing to the lack of technical maturity of this concept.

A contribution from the Republic of Korea presented the Pyro-processing Integrated Inactive Demonstration facility, which integrates the unit processes involved in the reprocessing of spent nuclear fuel by pyrotechnologies. The purpose of this facility is to perform integral tests of unit processes, evaluate safeguards issues, process optimization, providing scaled up information and training on pyroprocess facilities.

The main technology issues of the fast reactor fuel cycle have also been evaluated in the Russian Federation, where the objective is to improve safety, ecological acceptance and economic efficiency of fast reactor spent nuclear fuel reprocessing technologies. Technical goals include the ability to reprocess SNF with low cooling time and high burnup, non-proliferation requirements, Pu losses of less than 0.1%, end products directly suitable for fuel fabrication, low volume of high level wastes and partitioning. The analysis of these requirements results in the need for combined pyro- and hydro-reprocessing techniques, which are currently under development in the country. Activities in this area also focus on the treatment of high level waste arising from pyrochemical processes.

The evaluation of alternative fuel cycle options was discussed by a US representative; the main goal of this study being to provide a systematic, objective and traceable method for evaluating and categorizing nuclear energy systems. Technical details of the fabrication of fuel for the integrated recycling test were also presented, including casting and settler/bonder systems, an advanced remote welding system and specimens for irradiation testing. The integrated recycling test will be performed in conjunction with the fuel cycle technologies advanced fuel campaign irradiation test series.

The European project ASGARD (Advanced fuelS for Generation IV reactors: Reprocessing and Dissolution) was presented by Sweden. The objectives of the project, which will promote communication and cooperation among the different communities working on GEN IV concepts, focus on the behaviour of novel fuels ranging from production, conversion and refabrication of advanced fuels to education and training on nuclear materials from the whole fuel cycle.

EXPERIMENTAL TESTS, DATA AND ADVANCED SIMULATION

Track 7 of the FR13 Conference was devoted to Experimental Tests, Data and Advanced Simulation. Sixty-two papers were accepted for this track, by the track leaders Mr. A. Zaetta from France and Mr. Yu.S. Khomyakhov from the Russian Federation. Thirty papers were presented as oral presentations in one of the six technical sessions. The remaining 32 papers were presented as posters at the second poster session.

The contributions in this technical track covered the main problems and challenges related to advanced experimental tests and multi-physics simulations of fast reactor designs and safety analysis.

In the area of core neutronics, relevant efforts are devoted, for instance, to establishing standard practices and the means for uncertainty quantification in the neutronics codes used for the design and analysis of current and innovative fast reactor core designs, with particular attention to safety related aspects. The evaluation of calculation accuracy of sodium void reactivity effects, a key aspect for the design of new SFR concepts, the evolution of loss of flow scenarios and control rod withdrawal accidents can be mentioned as examples of these activities.

The improvement of performances and accuracy for thermohydraulics codes is also the purpose of numerous research projects carried out at national and international levels. Key issues are related, for instance, to the applicability of codes to liquid metals and non-conventional coolants. CFD modelling is currently very prominent, but several problems are related to the lack of sufficient quantification and qualifications of its uncertainties with available data. In this area, one of the main aims of research activities is the coupling of systems codes and CFD tools.

Similar efforts are also focused on fuel performance codes and structural mechanics simulation tools, which are fundamental to assessing the behaviour of fuels and materials in a fast reactor core environment.

The coupling of codes for multi-physics simulation represents one of the most challenging aspects in this field. Different code strategies for assessing the dynamic behaviour of reactor cores and for simulating postulated severe accident scenarios were presented during the technical sessions related to this track. An example is the FAST code developed at PSI, which consists of coupled neutronics (ERANOS and SERPENT), thermohydraulics (TRACE) and fuel behaviour (FRED) modules.

It was commonly shown that an improved understanding of the simulation and modelling methods is necessary in order to apply suitable safety analysis on postulated accident scenarios. However, even small scale efforts such as the development of control room simulations for the training of operators is evolving and delivers countable results for safety improvements.

In all of these fields, important benchmark activities at the international level are carried out under the aegis of the IAEA, NEA and GIF. Experimental activities are fundamental and will be of paramount importance in acquiring new data which can be used for validation, verification and qualification of computer codes.

FAST REACTOR DEPLOYMENT, SCENARIOS AND ECONOMICS

Several fast reactor deployment scenarios are being assessed worldwide, taking into consideration the main technical aspects of innovative fast reactors and related fuel cycles development, the major technical requirements and the different marked drivers, including resource utilization, fuel cycle options, waste management, economic competitiveness and proliferation issues. These efforts were discussed in three technical sessions devoted to the track on Fast Reactors Deployment, Scenarios and Economics, which included 16 oral presentations and 11 contributions in the form of posters.

Overall, the analyses recently carried out have highlighted the great potential and flexibility of fast neutron systems, which can be deployed as large size nuclear power stations for grid connection, small size plants for decentralized energy supply and non-electrical applications, as well as to implement different fuel cycle strategies (Pu management, minor actinide burning, etc.).

In France, a global assessment of fast reactor future deployment shows the need for this technology to achieve long term sustainability, although today large uncertainties make it difficult to assess the terms and timing of the realization of fast reactor fleets. Further multi-criteria analysis and investment scenario studies of fast reactors were conducted in order to identify favourable conditions of fast reactor deployment in the European market, also including considerations on non-electrical applications. The prospect of a closed Pu cycle from a utility's point of view was presented, discussing the various solutions needed to achieve the closure of the nuclear fuel cycle.

Criteria for fast reactor deployment assessment obviously vary from country to country. In India, long term energy security, mainly based on indigenous resources, is a priority which arises from economic, environmental and strategic considerations. In this context, the fast breeder reactor programme is considered essential for the energy security of the country, especially considering the possible utilization of thorium resources.

Deployment strategy studies after the Fukushima accident were discussed by a representative from Japan, where the future role of nuclear power is currently under discussion. The results of the analysis revealed a need for the implementation of reprocessing and the development of fast reactors in the medium to long term, in order to achieve the objectives for uranium utilization, spent fuel stockpile, Pu inventories and radioactive waste management.

Similar analysis has been carried out in the Russian Federation, where on the one side the technical maturity has been reached with BN series reactors and MOX fuels to face LWR spent nuclear fuel issues, and on the other side new technologies are being developed for long term deployment strategies. Several possible scenarios for BN-1200 MOX fuel reactors and the related spent nuclear fuel management after 2030 were recently assessed, including considerations on spent nuclear fuel storage time before reprocessing, proliferation resistance, costs and safety. Options for future small scale nuclear power productions are also being studied; in particular, small modular reactors with lead–bismuth coolant (SVBR-100) under development in the Russian Federation can represent a solution for future small and decentralized energy production.

Different prospects for fast reactor utilization are being explored in the USA. Although these systems were originally conceived to face concerns regarding scarcity of uranium resources, several key characteristics of innovative fast reactors are now being evaluated in order to reduce the capital and energy costs of these nuclear systems. These objectives are being pursued for instance by designing compact reactors which feature high thermal efficiency, extended core lifetimes and high fuel burnup.

In Europe, studies on fast neutron systems for minor actinide burning and recycling are also carried out within the 7th Euratom Framework programme and in several EU countries. Different options for Pu management in advanced SFR fuel cycles are being evaluated in the United Kingdom.

As for international initiatives, the main achievements of the IAEA INPRO collaborative project Global Architecture of Innovative Nuclear Energy Systems with Thermal and Fast Reactors and Closed Nuclear Fuel Cycle (GAINS) were presented.

FAST REACTOR OPERATION AND DECOMMISSIONING: INTERNATIONAL EXPERIENCE

Extensive experience in fast reactor design, construction and operation has been gathered worldwide during the past decades; the preservation of this knowledge and technology basis, gained mainly on SFR technology in France, India, Japan, the Russian Federation and the USA, is clearly recognized as representing a key element for the development and deployment of future fast neutron systems. These aspects were discussed in the two technical sessions devoted to the track on Fast Reactor Operation and Decommissioning: International Experience, which included 11 oral presentations and 8 poster contributions.

A representative from the French utility EDF gave a complete overview of fast reactor experience in France, covering major aspects from the design to the decommissioning of prototypes and industrial SFRs realized elsewhere. The main lessons learned from the choice of the design options, construction, operation and plant availability, accidents, fuel cycle management and decommissioning experience were summarized. In particular, the reprocessing and multiple recycle experiments performed in the Phénix reactor demonstrated the feasibility of the fast reactor closed fuel cycle, and it represents today a unique global resource for future developments. The Phénix and Superphénix projects also represent an important test platform for acquiring knowledge on dismantling of large components and decommissioning issues.

In India, the 26 years of successful operation of the FBTR represents the technology basis of the Indian fast breeder reactors programme. This experience provided fundamental knowledge on key aspects of SFRs such as fuel performance (including metallic fuels which are under development in the country), sodium systems and technology, instrumentation, structural materials, contamination aspects, and high pressure and high temperature steam generator operation. FBTR operation has provided sufficient confidence for the design and launch of the construction of the 500 MW(e) PFBR, currently in an advanced phase of construction at the Kalpakkam site.

In Japan, the restart of Monju, which reached criticality in 1994 and is one of the very few large size fast reactor plants operable today, is under discussion. Historical remarks, post-Fukushima safety measures and current status of the reactors were discussed by a Japanese representative. It was also underlined that the role of Monju as a prototype reactor continues to be fundamental, and some of the international joint research programmes using Monju are still actively being carried out. Monju is therefore expected to play a role as an international asset to provide a fast reactor research facility at power and a knowledge/technology transfer tool to future generations.

The Russian Federation is the country with the most experience in fast reactor technology. The largest SFR in the country is the Beloyarsk power unit No. 3, which is equipped with a BN-600 reactor, commissioned in April 1980 and is currently under routine operation. This 33 years of operating experience is positive in terms of the demonstration of the feasibility of the utilization of SFRs for commercial electricity generation. Further, the reactor is a key link for ensuring the continuity of the fast reactor programme in the country, which envisages the construction of the BN-800 nuclear power plant and the design of the BN-1200. Papers detailing experience and solutions on the decommissioning of SFRs, which are also an important part of the extensive Russian experience on SFRs, were presented.

Important experience in fast reactors was also gathered in the USA, in particular with the EBR-II reactor which represented a major contribution to the development of SFR technology. The reactor was operated for 30 years and allowed advancing SFR technology in several aspects, e.g. metal and oxide fast reactor fuels development, operational safety tests which demonstrated the self-protecting nature of fast reactors and fuel recycle technology. Decommissioning aspects were also extensively explored through the EBR-II, which was successfully decommissioned after the 30 years of operation.

SKILL CAPABILITIES, PROFESSIONAL DEVELOPMENT, KNOWLEDGE MANAGEMENT

National and international activities aimed at the management of knowledge and education and training in the field of fast reactors and related fuel cycles were presented during the two technical sessions held on the track on Skill Capabilities, Professional Development, Knowledge Management, which overall included 11 oral presentations.

In France, significant efforts are currently devoted to the preservation of the knowledge acquired in the past decades in the country, in particular through the development, design and operation of a number of SFRs. This know-how is considered a key element in the perspective of the realization of the GEN-IV ASTRID prototype. As for education and training, the continuous operation of the French Sodium School and of the Phénix plant simulator have created a favourable context to restart education and training courses and tools in the field of SFRs. Besides the national context, the representative from France gave an overview of the initiatives around the world aimed at the education and training in support of the SFR programme, which are currently increasing due to the greater interest in this technology.

Another important project with education and training purposes is ELECTRA, a low power (0.5 MW) LFR facility under development in Sweden. The reactor will be used as a training facility for GEN-IV LFR and SFR systems, as well as a test platform for LFR technology development, research on fast reactor core physics and dynamics, R&D on fuel recycle and manufacture. At the European level, another relevant initiative in education and training is the ENEN-III project, which contributes to promoting education and training in the area of GEN-IV systems.

International collaboration is recognized as being fundamental in this area. An example of a cooperative project is the Information Sharing Framework for Facilitating Development of Fast Reactors and Fuel Cycles, under development by the Japan Atomic Energy Agency, Sandia National Laboratory, the Korea Institute of Nuclear Non-proliferation and Control and the Korea Atomic Energy Research Institute. The objective of the project is to promote sharing of expertise in order to support nuclear transparency and facilitate the development of fast neutron reactors.

Other international activities in this area are carried out under the aegis of the IAEA, which has developed a specific digital archive — the Fast Reactor Knowledge Organization System — which was presented ‘live’ at the conference. This is a repository of publications organized and searchable online by approved and registered users. It uses a formal taxonomy developed by the IAEA specifically for fast reactor technology. Further IAEA activities in the area of fast reactor knowledge preservation were also reviewed during the technical sessions.

THE YOUNG GENERATION EVENT

For the second time after the FR09 conference, the IAEA staged a Young Generation Event at the FR13 conference dedicated to young professionals working in the different fields of fast reactors and related fuel cycles.

The event was based on a web video contest launched by the IAEA in the forefront of the conference. Young academics and professionals under the age of 35 were invited to submit a brief video introducing a project in the field of fast reactors and related fuel cycles. Five exceptional videos were selected and presented during the conference. The Young Generation Event included a first workshop (6 March 2013) to identify and elaborate current trends and challenges in the fast reactors and related fuel cycles field. The following six main areas were identified:

- Sustainability
- Innovation
- Simulation
- Safety
- Economics
- Public acceptance

For each of these areas, a working group elaborated definitions, current challenges and interlinking synergies. Finally, each group nominated a spokesperson who presented their results at the following Young Generation Event. The second step was the final Young Generation Event held on Thursday, 7 March 2013, in the main hall of the conference premises. Around 150 people made their way to the event to listen to the ideas and visions of the young generation. The six nominated spokespersons tried jointly to find acceptable solutions to ensure sustainability and energy supply for the future. Overall, the idea of the Young Generation Event was to create a bridge between the visions of young academics and experienced scientists in the field of fast reactors and related fuel cycle technologies. This idea developed into a fruitful panel discussion, which reflected the widespread awareness of young professionals in matters of technology development, safety aspects, economic dependency and environmental responsibility.

CLOSING SESSION

In the closing session of the conference, an overview of the technical sessions was provided by Mr. F. Carre from the CEA, France, who remarked that research and projects on fast neutron reactors and related fuel cycles remain at a sustained level worldwide. Emphasis has recently been put on safety aspects, in the aftermath of the accident occurring at the Fukushima Daiichi nuclear power plant; a relevant initiative on this aspect is the establishment of internationally agreed safety design criteria for GEN-IV SFR systems. Diversities and complementarities of national projects were also highlighted, from near term SFR programmes to long term projects, which also include promising innovations not only for SFRs but also for LFRs, GFRs and molten salt fast reactors. After a review of the main highlights and outcomes of the technical sessions, Mr. Carre concluded his speech by expressing his appreciation for the active participation at the oral and poster sessions and thanked all the conference organizers.

Pathways to the deployment of GEN-IV reactors were discussed in the concluding remarks given by Mr. C. Behar from the CEA, France, who underlined the current efforts to define and meet GEN-IV objectives, e.g. GIF roadmap update, GIF safety design criteria for SFRs, feedback experience from the Fukushima Daiichi accident, Western European Nuclear Regulators Association and IAEA safety standards. Examples of recent promising results and innovations presented at the conference were also provided.

An invitation to the next IAEA international conference on Fast Reactors and Related Fuel Cycle — FR17 was provided by Mr. V. Rachkov from IPPE, Russian Federation, on behalf of Mr. V. Pershukov from ROSATOM. The FR17 conference is to be held in St. Petersburg in 2017.

The final closing speech was given by Mr. S. Monti, Scientific Secretary of FR13, who remarked on the impressive participation at the conference and the excellent quality of the scientific contributions. Closing his speech, gratitude was expressed in particular to the International Advisory Committee which provided advice on the scope, overall objectives and structure of the conference, and to the International Scientific Programme Committee, which set up the detailed conference programme, identified key speakers and, last but not least, selected and peer reviewed almost 400 scientific contributions. The joint efforts of the chairpersons, the International Advisory Committee, the International Scientific Programme Committee, the Local Organizational Committee and the Conference Secretariat were fundamental for making the FR13 conference a success.

OPENING SESSION

Chairpersons

C. BEHAR
France

S. MONTI
IAEA

OPENING ADDRESS

L. Michel
French Ministry of Ecology, Sustainable Development and Energy,
Paris, France

Good morning, Ladies and Gentlemen

As a representative of the French Ministry of Ecology, Sustainable Development and Energy, I am honoured to deliver an opening address to the 2013 International Conference on Fast Reactors and Related Fuel Cycles (FR13), organized by the IAEA and hosted by the Government of France through the French Alternative Energies and Atomic Energy Commission (CEA) and the French Nuclear Energy Society (SFEN).

First of all, I would like to express my appreciation that so many participants, both from home and abroad, are attending this conference. I'm also 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 conference: FR13. For this conference, about 600 participants have registered from various countries and three international organizations (the European Commission, the OECD Nuclear Energy Agency, and the IAEA). I am very impressed by the quality of the numerous papers submitted to this conference.

For this opening address, I would like to share with you some thoughts about the evolution of the key drivers during the last decades for the development of fast reactors from the very pioneering age until now, taking into account new concerns and the major events that have occurred since the last international conference on fast reactors and related fuel cycles held in 2009 in Kyoto, Japan (FR09).

There are three major periods:

— The pioneering age (1945–1980) with breeding as a main driver followed by a kind of “winter season” (1980–2000) for the development of fast reactors worldwide.
— The so-called “brainstorming” phase (2000–2010), back to physics and nuclear chemistry, with international rebirth of the research on fast reactors and advanced fuel cycle owing to the GENERATION IV initiative, revisiting various reactor concepts along with 4 main drivers: sustainability, safety, proliferation resistance and cost competitiveness.

— The new era (started in 2010) with very promising technological options and projects of prototypes with two main key drivers:
  - Innovation towards enhanced safety, which is a major concern for public acceptance of nuclear power, especially after the Fukushima accident;
  - Higher flexibility in the management of fissile materials and nuclear waste in order to take into account various possible options for the contribution of nuclear power in the energy mix.

Let me briefly elaborate on these three periods.

— The pioneering age

As you may know, the story started with the brilliant intuition of talented physicists. Among them, one can mention Enrico Fermi who stated in 1945 that "The country which first develops a breeder reactor will have a great competitive advantage in atomic energy".
Based on this principle, nearly 62 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. In the following two decades, several countries followed the US in launching an intensive fast reactor programme. Plutonium fuelled breeder reactors appeared to offer a way to avoid a potential shortage of the low cost uranium required to support such an ambitious vision using other kinds of reactors.
As you know, the development of fast reactors worldwide was however not so rapid, mainly for three reasons. Firstly, proliferation concern emerged in the seventies when the Ford and Carter Administrations decided to oppose further export of reprocessing technology. The 1978 Nuclear Non-proliferation Act (NNPA) sought to tighten the criteria for nuclear cooperation and reshape the nuclear fuel cycle. Secondly, uranium proved to be much more abundant than originally imagined and, after a rapid start, nuclear power growth was slower than projected in the early 1970s. It was also due to the fact that the 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. Thirdly, because of the high costs, and reliability and safety issues, no commercial breeder reactors have been deployed.

It took about 20 years to realize that nuclear energy could expand again, owing to the energy and climate challenges the world was faced with, and with that, the potential use of fast reactors became possible in order to account for the constraints of such expansion.

— The so-called "brainstorming" phase (2000–2010) towards the rebirth of fast reactors owing to the GENERATION IV international initiative

In 2000, the context was quite different and the development of fast reactors had to be made on a new basis, taking into account new drivers such as cost issues, safety and reliability, sustainability (resource saving and 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.
A variety of fast reactors can be considered for transmutation. The fundamentals of fast reactors were therefore revisited using different coolants (sodium, lead or gas), and much effort was also devoted to study the feasibility of various advanced fuel cycle options to separate and burn some of the minor actinides such as neptunium, americium, or curium.
This GENERATION IV programme led to various incentives worldwide to study the feasibility of technological demonstrators of advanced fast reactors. Consider the programme launched in Europe.

— The new era

A new era with very promising technological options and projects of technological demonstrators includes the case of Europe and the example of France. As you may know, the European Council committed in March 2007 to very ambitious goals putting Europe at the forefront of the fight against climate change, and launching the Strategic Energy Technology (SET) Plan, which identifies a list of competitive low carbon energy technologies to be developed and deployed in Europe. Nuclear power was of course included in this list.

In that respect, the European Sustainable Nuclear Industrial Initiative (ESNII) was officially launched in November 2010 under the SET Plan. Along with GENERATION IV key objectives, ESNII promotes advanced fast reactors with the objective of enhanced safety, resource preservation and minimization of the burden of radioactive waste for a more sustainable development of nuclear energy.
The main objective of ESNII is to maintain European leadership in fast spectrum reactor technologies.
With respect to the 2010 evaluation of technologies, sodium is still considered to be the reference technology since it has more substantial technological and reactor operations feedback.
The lead(-bismuth) fast reactor technology has significantly extended its technological base and can be considered as the shorter term alternative technology, whereas the gas fast reactor technology has to be considered as a longer term alternative option. Therefore, ESNII is aiming at promoting a consistent programme based on two main European projects: the Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID, French project) and the flexible fast spectrum irradiation facility MYRRHA (Belgian project). At this stage of my talk, I would like to focus a little bit on the R&D undertaken in France for fast systems and the related fuel cycle.
Research activities in this field are implemented according to two French Parliamentary Acts, i.e. the 13 July 2005 Act specifying the energy policy guidelines and the June 2006 Act outlining policies for the sustainable management of radioactive materials and waste. In the latter act, a major milestone was defined for the end of 2012 and a status report was sent to French authorities reviewing the main results obtained since 2006. This very valuable work is of major interest at a moment when we have just launched a new national public debate about the future of our energy policy. This report has pointed out some key issues that I would like to share with you.
The first issue is of course public acceptance of nuclear power especially after the Fukushima accident, requiring higher safety standards and technological innovations. In that respect, this report is reflecting major achievements to overcome by design some weaknesses of past generations of sodium fast reactors, improving the natural behaviour of the core (core with negative sodium void coefficient), taking into account severe accidents in the design stage to implement mitigation strategies (core catcher), investigating innovative power conversion systems to avoid (or to minimize the risks associated with) sodium reactions with air or water. The second issue is dealing with the sustainable management of fissile materials and nuclear waste. Whatever the energy mix chosen, we will have to deal with

the plutonium build-up in PWR spent fuels and with the management of the long lived radioactive waste. In that respect, fast reactors and associated closed fuel options have demonstrated very flexible features.

In summary, we know now that Fermi's initial intuition is confirmed. Promising innovation routes are now clearly identified to further enhance safety, reduce capital cost and improve efficiency, reliability and operability, making the Generation IV sodium fast reactor concept an attractive option for electricity production. We thus rely on you scientists to demonstrate that these innovations are robust and to convince our stakeholders to go further. Such a conference is a unique opportunity to discuss significant issues on fast reactor development and related fuel cycle development.
I sincerely wish that this conference will lead to a fruitful debate and that international cooperation dealing with research and development of fast reactors will be further strengthened.

Thank you very much for your attention.

OPENING ADDRESS

Y. Amano
International Atomic Energy Agency,
Vienna

I am pleased to address this opening session of the International Conference on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios, organized by the IAEA and hosted by the Government of France through the French Alternative Energies and Atomic Energy Commission (Commissariat à l'énergie atomique et aux energies alternatives).

This event follows the successful conference in Kyoto, Japan, in 2009, which I also had the honour to address. Since then, the nuclear world has been shaken by the serious accident at the Fukushima Daiichi plant in Japan, two years ago this month. Despite some predictions to the contrary, global use of nuclear power will continue to grow in the next few decades. Major existing users such as China and India have significant expansion plans. Nearly a dozen countries, both developed and developing, plan to build their first power reactors. The United Arab Emirates last year became the first new country for 27 years to start construction work on a reactor.

But the Fukushima Daiichi accident has left an important legacy, which is a much stronger global focus on safety. I am therefore pleased to see that this conference addresses the issues of safety and sustainability. Public confidence in nuclear power was greatly shaken by the Fukushima Daiichi accident. It will take time to rebuild that confidence. This will only be possible if everyone involved in nuclear power has a total commitment to safety and if the sector is open and transparent. The public needs to be reassured that nuclear energy is efficient and safe, can mitigate the effects of climate change and can play a key role in meeting the growing global demand for energy. Fast reactors and related fuel cycles will be important for the long term sustainability of nuclear power. This innovative technology has the potential to ensure that energy resources which would run out in a few hundred years, using today's technology, will actually last several thousand years. Fast reactors also reduce the volume and toxicity of the final waste.

China's Experimental Fast Reactor has been connected to the grid. Work is at an advanced stage on construction of India's 500 MW(e) Prototype Fast Breeder Reactor and of the large BN-800 reactor in the Russian Federation. Interest in fast reactors with closed fuel cycles is increasing steadily. A number of emerging economies are joining the existing fast reactor technology holders.

Considerable R&D work is being done on advanced designs with enhanced safety characteristics. It is important to gather the operational experience of countries with operating fast reactors and related fuel cycle facilities. This can help to achieve higher levels of safety. Events such as the Joint GIF-IAEA Workshop on the safety of sodium cooled fast reactors last week are a useful way of doing this. They also help to ensure that relevant lessons from the Fukushima Daiichi accident are learned.

The IAEA remains the unique collaboration forum for ensuring continued progress in fast reactor technology. We provide an umbrella for knowledge preservation, information exchange and collaborative R&D in which resources and expertise are pooled.

Let me conclude by thanking all of the experts in the International Advisory Committee, the International Scientific Programme Committee and the Local Organizing Committee — as well as my IAEA colleagues — who have worked so hard in the last year to organize this conference.

I wish you every success with this important event.

Thank you.

OPENING ADDRESS

B. Bigot
Commissariat à l'énergie atomique et aux énergies alternatives,
Saclay, France

Mister Chairman, Ladies and Gentlemen,

It's a great pleasure and an honor for me to attend the opening ceremony of the international conference on fast reactors and related fuel cycles under the auspices of the International Atomic Energy Agency here in Paris. I thank the Organizing Committee for having invited me to present an overview of the challenges we have to face in order to elaborate a long term sustainable energy policy while coping with the short term constraints, not only for France or Europe, but more broadly worldwide. Indeed, I am very pleased to share with you my deep belief that the fast neutron reactors and the good complementarity between nuclear and renewable energies represent key assets for such a policy.

The world has to deal with an unavoidable increase in energy needs to satisfy the legitimate expected social and economic development of many countries in the next two decades, despite all the efforts we have to do for better energy efficiency and for large energy savings in the most advanced economies.

In Europe today, the energy supply relies on fossil fuels for over 75% of its primary energy consumption. Worldwide, it is over 82%. Such a massive use of fossil fuels is thus both a threat for the environment, climate and health, and also for our economies. In 2012, for France, fossil fuel imports represented an expense three times larger than in 2005, and accounted for over 90% of France's trade deficit.

If keeping this current energy mix, we will strongly contribute to a large increase of the risk of climate change, environmental and human health impacts, and their damaging effects. Tackling climate change, environmental and health issues will require the priority use of CO_2 emission free energy sources. Despite the Kyoto Protocol and many political international statements, the amount of CO_2 emissions per year has known a +40% growth from 1990 to 2009, with the correlated increase of temperatures. To ensure a sustainable development, the world needs a sustainable energy supply, which makes a sufficient amount available for everybody at an acceptable price. For all these reasons, the substitution of fossil fuel consumption as soon as possible and to the greatest extent possible with CO_2 free energy sources must be our top priority. The cornerstone of any sustainable European energy is the reduction of our consumption of fossil fuels with three axes of action relative to the technologies using these fuels: energy savings, improved efficiency and substitution by other technologies which do not use such fuels, such as renewable and nuclear energies.

In November 2011, the European Commission initiated a project entitled Energy Roadmap 2050, with more ambitious targets than those of the 3x20 objectives for 2020. All studied scenarios in the frame of the roadmap include an increase of the share of the renewable energy sources, a decrease in the use of fossil fuels, a contribution of the nuclear technology in the range up to 20%, depending on the hypothetical public acceptability, and a large improvement in energy efficiency. The renewable energy sources will require specific developments in terms of grids, storage capacities, and production and connection infrastructures. According to this scenario and to cope with the intermittency issues, gas will very likely remain a significant energy source in the European mix provided we develop carbon capture and sequestration or recycling. For France, we believe we may be even more ambitious and rely only on nuclear and renewable energies by 2050. In the CEA, we are working in all carbon free energy sectors with one major goal: to decarbonize the energy production in France completely. High uncertainties are attached to the proposed scenarios, particularly regarding energy sources and energy storage. The competitiveness of European industrial companies as well as the buying power of the consumer needs to be preserved and the environmental constraints must be accounted for. In these conditions, it is of first importance to keep open a reasonable spectrum of options, to have a pragmatic strategy not to substitute a technology by another at large scale before having demonstrated the real capacity of this new one in terms of reliability, integration and economy, and to develop synergies between the various carbon free energy sources. Indeed, nuclear energy and renewable energies have a large potential for complementarity. Nuclear is a massive, concentrated, continuous and as steady as possible way of producing electricity. Furthermore, it produces a large quantity of fatal heat, a part of which could be used. Renewable energies are diffuse, low intensity, intermittent ways to produce electricity, heat or mechanical forces, and mostly non-programmable ways with an acceptable reliability of more than a few hours. Most of them (waterpower, photovoltaic, windmills, the various marine energies, and geothermal energies) however produce electricity. Nuclear is well suited for the electricity base load needs while renewable energies are acceptable for more flexible uses if we may develop suited means of short lasting (a few hours or a few days) energy storage. Indeed, when we speak about renewable energy, since we have local production, we must favour local storage and consumption to avoid the grid bottlenecks and unreasonable overinvestment costs. Worldwide, it is foreseeable that nuclear, which is a carbon free energy and has key advantages for security of supply and economic competitiveness, will keep a significant place while the share of renewables will increase.

In spite of the Fukushima accident, nuclear energy remains a widely shared option worldwide, even if some countries decide or consider a nuclear renouncement or a phase out. Actually, nuclear energy is an energy that will grow in the world and many countries confirmed their nuclear option, because they need it: Brazil, China, Finland, India, Republic of Korea, Poland, the Russian Federation, Saudi Arabia, South Africa, Turkey, the United Arab Emirates and others, need to utilize this option to meet their rapidly growing electric and drinking water demands. For them, nuclear is not an option, it is a necessity. It was a necessity before Fukushima; it is still a necessity after Fukushima. What is no longer an option is the strict respect for the highest international rules and standards for safety and design of the new reactors and for operation, even in the most unlikely events. Some 60 new reactors are currently under construction in the world. Nuclear is and will still be a pillar of France's energy policy. It allows France to produce four times less CO_2 emissions for electricity production than all other large European countries and to benefit from an electricity less than half as expensive as in some other European countries. The goal fixed by President Holland to pass from over 75% of our electricity supply coming from nuclear as it has been for the past 20 years to 50% in the next 20 or 30 years is consistent with the substitution of fossil fuel consumption with renewable energies. Now, fossil fuel usage accounts for 43% in the housing and service industries and for 31% in ground transportation. The renewables produced mainly electricity and, in a less proportion, heat. Both of them could be advantageously used in the housing and service industries and in ground transportation. So the share of the electricity in our energy consumption will increase and so, even if we keep steady our nuclear capacity, the nuclear share will decrease. The priority of France's energy policy is a reduction of our fossil fuel consumption, not a reduction of our nuclear capacity by substituting it with extra fossil fuels capacities as some countries are doing when they phase out nuclear.

After Fukushima, it is really essential for nuclear energy to regain confidence, locally in the vicinity of the nuclear power plants, at the level of each country, and on the international scene. Confidence of the public in our long term energy policy is absolutely needed. This is not a partisan issue; in any country, a large majority should adopt it. If there is no stable consensual policy and regulations, nobody will invest in nuclear, a technology requiring a highly intensive capital expenditure but with a low operational cost. It is necessary to raise the safety standards and to guarantee their widest application through the international community. The priority must be given to the no-release of damaging radioactive materials outside the nuclear site, whatever the normal, extreme natural conditions or accidental situations. The generalization of the 'stress tests', the peer reviews, the promotion of the best technical standards, are some of the many initiatives aimed at fulfilling that need. The question of crisis management at the international level must be also addressed. But beyond each specific provision implemented in each specific country, it is also very important for confidence to build common rationales and shared practices on nuclear safety at the international level. Fukushima has shown once again that a nuclear accident somewhere has an impact everywhere. International cooperation and the multilateral bodies play a critical role in this respect.

The fuel cycle and back end management will remain another pillar of the confidence necessary for the use of nuclear energy in the long term. Each country using nuclear energy should have a sound and comprehensive fuel cycle policy. Of course, various policy options such as open or closed fuel cycles may be considered in different countries. France chose the closed fuel cycle, recycling the uranium and plutonium present in the nuclear spent fuel and allowing dividing waste volume and toxicity respectively by a factor of five and ten.

Expected nuclear growth in the world will not be possible without sustainability. For sustainable nuclear energy, research on future nuclear reactors is essential in order to design reactors known as IVth generation that could ensure power production for hundreds of years with a much better use of natural resources (a hundred times increase in the energy extraction from the uranium ores compared to the present technology), more environmental friendly techniques for extraction of uranium and a high level waste reduction. The CEA is working in order to define an innovative sodium cooled fast neutron prototype, called ASTRID, in the framework of the 2006 French Act on sustainable management of radioactive materials and wastes. The CEA developed international R&D partnerships with many countries or organizations: for example, with Japan on core degradation, or on optimization of reprocessing plants for fast reactor fuels; with India, on safety and severe accidents; with the Russian Federation on core physics; in the framework of GIF, on safety, etc. Any energy policy should consider the long term perspective. For example, designing and constructing a new type of nuclear reactor will require at least 20 years with a follow-up operational time lasting a minimum of 60 years. This means that what we are doing today will have an impact on our energy mix up to the end of this century.

The research at CEA is performed along three timelines:

(1) Short term: Support the nuclear industry through the improvement of our present nuclear fleet (Generation II).
(2) Medium term: Enhancing nuclear safety by slashing the risks of new nuclear build (Generation III).
(3) Long term: Pave the road to establish a sustainable nuclear future through the ASTRID project (Generation IV).

We are also preparing the long term future through partnerships, gathering European partners and six other countries, implementing a considerable research programme on controlled magnetic fusion and the ITER project, which aims to demonstrate the extraordinary potential of nuclear fusion energy.

In 2013, we will see in France the organization of two major debates:

- One will be on the radioactive waste deep disposal project, with the objective that the repository will be in operation by 2025.
- The other will be on energy transition. The French government is currently managing this energy debate for the next six months, with the objective to have a planning Act for energy transition by the end of the first semester 2013, including diversification of energy sources and promotion of energy efficiency.

The energy transition is necessary and the CEA, with its skills, wishes to be instrumental to this energy debate and is participating through the French National Alliance for Energy Research Coordination (ANCRE). The French Minister of Ecology, Sustainable Development and Energy, asked ANCRE to give its perspective on the possible ‘future’ for France, in a European and global context, by focusing her approach on scientific and technological aspects, including the potential associated with breakthroughs and technological innovations. ANCRE decided to propose 3 scenarios of French energy mix evolution at the horizons 2025 and 2050: termed ‘Maximum sobriety’, ‘Drastic reduction of fossil fuel dependence by a priority use of electricity’ and ‘Diversified supplies to reduce fossil fuel dependency’.

To decrease dependence on fossil fuels, France, like some other countries, has decided to build its mix on nuclear and renewable, with the use of fossil fuels reduced as soon as possible, and to prepare energy transition towards a carbon free energy mix. The optimum combined uses of nuclear electricity (as steady as possible with a time scale for significant production change of the order of the day) and renewable (time scale for significant production change of the order or a fraction of an hour) requests a large capacity of energy storage, which must be drastically developed — these energies are intermittent (no solar energy during the night, no wind energy without wind). Research on storage remains essential. Renewable energies also require development of communication and information technologies to the service of the electrical grids: these smart grids will be able to match supply and demand. The modelling of electricity production from renewable sources is also important and will enable anticipation of intermittent production. It is only in a diversified mix that we will find the guarantee of our energy independence. It is by abolishing the barriers between energy production and uses that we will be able to think about original solutions for greater energy efficiency. We have to achieve a global approach of energetic systems and the CEA is exactly working according to this model, from basic research to technological research up to the demonstrator.

In conclusion, I would like to share with you my deep conviction that fission nuclear energy, with a mastered and always improved safety, with a high safety culture of all the stakeholders and in a first row the operators and nuclear control authorities, will go on playing a major role in the 21st century and beyond since it answers the criteria of sustainable development. It will contribute in an important way to the world energy needs in complying with the expectation of our fellow citizens for low impacts of energy uses on climate, environment and health.

With the help of major international collaboration, the CEA is committed to demonstrate pathways for making significant progress for future fast reactors and their corresponding fuel cycle.

In synergy with nuclear energy, the renewable energies must be widely developed.

Thank you for your attention.

FAST REACTOR DEVELOPMENT AND WORLDWIDE COOPERATION IN THE GENERATION-IV INTERNATIONAL FORUM

Y. SAGAYAMA
Former GIF Chair,
Ministry of Education, Culture, Sports,
Science and Technology of Japan,
Tokyo, Japan

OVERVIEW

The Generation-IV International Forum (GIF) is a framework for international cooperative activities on research and development (R&D) of Generation-IV (Gen-IV) reactors, including fast reactors. This paper introduces its background, goals, activities, recent topics, such as formulation of draft safety design criteria (SDC) for the sodium cooled fast reactor (SFR) and the second GIF Symposium held in San Diego, the United States of America, in November 2012, as well as the future direction of GIF.

1. BACKGROUND OF GIF

GIF started more than 10 years ago. Looking back to the history of advanced reactor development in the world in the end of the 20th century, we found stagnation of nuclear R&D stemming from such 'headwinds' as budget cuts in nuclear programmes in the USA, the decision on the decommissioning of Superphenix in France, and the accident in Monju in Japan.

Breaking down the situation, the USA convened an international workshop in January 2000 to discuss international collaboration for promoting and increasing efficiency of the development of Gen-IV nuclear energy systems, resulting in the establishment of GIF. The GIF Charter was signed by Argentina, Brazil, Canada, France, Japan, Republic of Korea, South Africa, the United Kingdom and the USA. Subsequently, it was signed by Switzerland in 2002, the European Union in 2003, and China and the Russian Federation, both in 2006.

2. BASIC CONCEPT OF GIF ACTIVITIES

It is a common wish for everyone in the world to ensure the peaceful use of nuclear energy, safety, and non-proliferation and nuclear security. Meanwhile, huge investment is required for new reactor development. In order to allow Member States to advance cooperative R&D reasonably and efficiently, we defined technology goals for Gen-IV systems and selected six most promising systems for commercial deployment by around 2030, resulting in the formulation of the Technology Roadmap for Generation IV Nuclear Energy Systems, which identifies associated R&D.

Four technology goals for the Gen-IV systems is to provide significant improvements in sustainability, economics, safety and reliability, and proliferation resistance and physical protection compared with Generation-III and other energy systems. Sustainability goals consider efficient fuel utilization and waste minimization. In this regard, fast reactors have the potential to efficiently utilize uranium resources and to reduce the volume and radiotoxicity of high level waste by burning minor actinides (MA) and other radioactive nuclides. Economic goals are defined with focus on life-cycle cost and investment risk, and safety and reliability goals consider such systems that a core disruptive accident is unlikely to happen and no off-site emergency response is required even if a core disruptive accident is assumed. Proliferation resistance and physical protection consider the means for securing prevention of nuclear material from military use and for increasing resistance against terrorism.

We are promoting cooperative activities by launching projects on R&D themes common to various design concepts in each system, with the aim of establishing technologies that can achieve these goals.

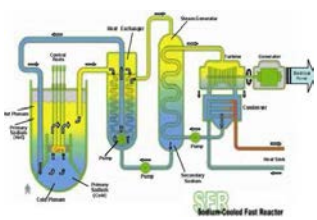
3. HISTORY OF GIF ACTIVITIES

Among the signatories to the Charter, 10 members have signed the Framework Agreement (FA) serving as the basis for their participation in GIF activities. Each party to the FA designates implementing agents to undertake cooperative R&D activities for several projects such as material, fuel, etc., by signing system arrangements (SA) shown in Fig.1. Memoranda of Understanding (MOU) for SAs were signed in 2010 for the Molten Salt Reactor (MSR) and the Lead-cooled Fast Reactor (LFR) systems.

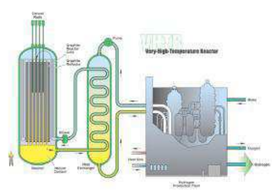
GIF members identified six reactor systems from among over 130 candidates as the Gen-IV reactors (see Fig. 2). The systems selected for development are the Very High Temperature Reactor (VHTR), SFR, LFR, Gas-cooled Fast Reactor (GFR), Supercritical Water-cooled Reactor (SCWR) and MSR. Among the six systems, the SFR, GFR and LFR use only fast neutron and the SCWR and MSR can use both thermal neutron and fast neutron.

	CAN	FRA	JPN	ROK	ZAF	CHE	USA	EUR	PRC	RUS	UK	RFB	ARG
Sodium-cooled Fast Reactor (SFR)		X	X	X			X	X	X	X			
Very High Temperature Reactor (VHTR)		X	X	X		X	X	X	X				
Gas-cooled Fast Reactor (GFR)		X	X			X		X					
Supercritical Water-cooled Reactor (SCWR)	X		X					X		X			
Lead-cooled Fast Reactor (LFR)			MOU					MOU		MOU			
Molten Salt Reactor (MSR)		MOU						MOU					

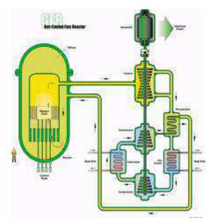
FIG. 1. Membership to GIF system arrangements.



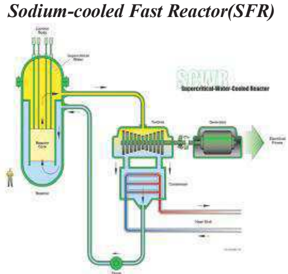
Sodium-cooled Fast Reactor(SFR)



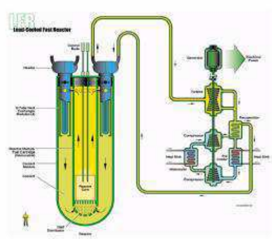
Very High Temperature Reactor (VHTR)



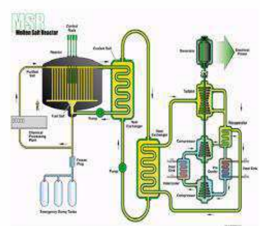
Gas-cooled Fast Reactor (GFR)



Supercritical Water-cooled Reactor (SCWR)



Lead-cooled Fast Reactor (LFR)



Molten Salt Reactor (MSR)

FIG. 2. Six Gen-IV systems.

The Gen-IV Technology Roadmap was issued in 2002, which covers overall prospective on milestones and budgets necessary for future R&D by fully examining R&D plans of Member States on the six nuclear systems.

The GIF comprises the Policy Group (PG) as the highest policy-making body, System Steering Committees (SSC) responsible for managing cooperative activities for each system and project, working groups (WGs) working on the development of evaluation methodology, and a board of Senior Industry Advisory Panel (SIAP) providing advice from the nuclear industries (see Fig. 3).

For WGs on the evaluation methodology, the Economic Modelling WG, the Proliferation Resistance and Physical Protection WG and the Risk and Safety WG have established methodologies for evaluating progress of each system against the technology goals.

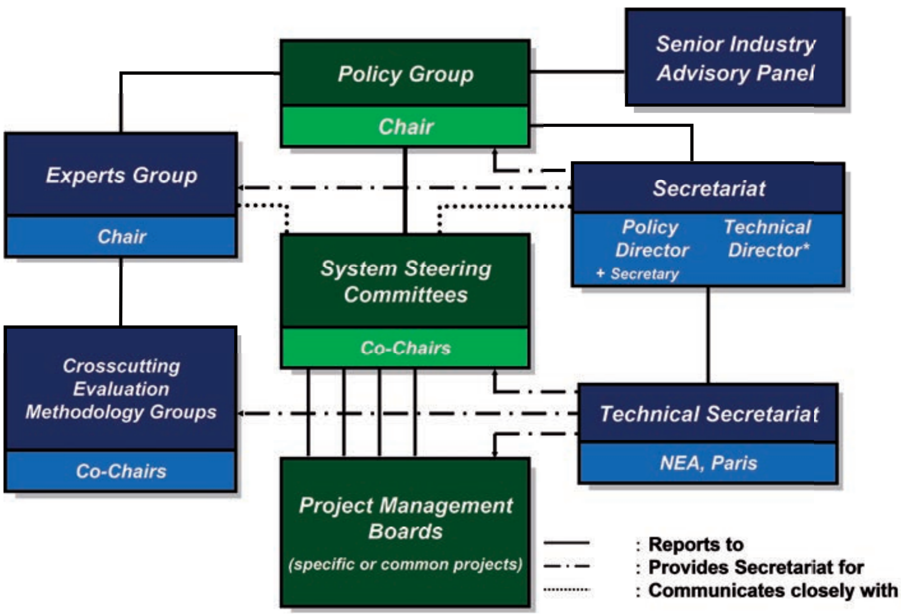


FIG. 3. GIF governance structure.

4. CURRENT STATUS OF GIF ACTIVITIES

Cooperative activities on fast reactor systems, recent topics and changes in circumstances surrounding GIF include:

— Cooperative activities on fast reactor systems:
  - SFR: Four cooperative R&D projects, Advanced Fuel (AF), Component Design and Balance-of-Plant (CDBOP), Safety and Operation (SO), and Global Actinide Cycle International Demonstration (GACID), are being actively carried out. Project agreement (PA) for a new project, System Integration and Arrangement (SIA), is awaiting signing. The SIA project is expected to conduct comprehensive evaluation of R&D issues. Currently, China and the Russian Federation, which own SFRs, are in the signing process for participation in the projects and thereby more vigorous cooperation is expected. The four reference reactors for SFR are loop-type JSFR (Japan), pool-type KALIMER (Republic of Korea), ESFR (EU) and small modular SFR, SMFR (USA).
  - GFR: For the GFR, cooperative R&D on decay heat removal in the Conceptual Design and Safety (CD&S) project is in progress. The Fuel and Core Materials (FCM) project is under preparation. Meanwhile, a collaborative project on an experimental reactor, ALLEGRO (75 MW(e)), started up in Europe in 2010.
  - LFR: The LFR has been developed since the 1960s to serve as a nuclear submarine reactor in the Russian Federation. Aiming at design with a high margin of safety, taking advantage of its features such as the high boiling temperature of lead (1790°C), R&D activities are being conducted on the issues regarding oxygen control, earthquake-proof, and in-service inspection (ISI). The reference reactors are the ELFR (EU, 600 MW(e)), the BREST-300 (Russian Federation, 300 MW(e)) and the SSTR (USA, 20 MW(e)). Conclusion of SA is expected.
  - SCWR: The SCWR is a high temperature, high pressure water cooled reactor that incorporates technologies of Gen-III^{+} and the advanced fossil fuel power plant that operates above the thermodynamic critical point (374°C, 22.1 MPa) of water. There are two main types: pressure vessel concepts and pressure tube concepts, and it can be built as a fast reactor. Materials and chemistry projects and thermohydraulic and safety projects are ongoing.
  - MSR: MSRs developed in the USA in the 1960s and 1970s were thermal neutron spectrum reactors, but recent R&D within GIF has focused on fast spectrum MSR concepts. The reference concept is the molten salt fast reactor (MSFR) being developed by France and Euratom. SA on the MSR is expected to be concluded.

— Recent topics
- Development of Safety Design Criteria (SDC): The development of SDC for the SFR, which serve as international safety criteria common to SFR systems, was proposed in 2010 as the first step for SDC harmonization for Gen-IV systems. A task force (TF) team, comprising experts on the SFR and safety, started to develop the SFR/SDC in 2011 and made out a draft SDC at the end of 2012. It is based on the IAEA's standard for the LWR and includes safety issues that are specific to the SFR and lessons learned from the TEPCO Fukushima Daiichi Nuclear Power Plants accident, as shown in Fig. 4. In order for this SFR/SDC to be applied as an international guideline in countries including non-members of GIF, we strive to cooperate with the IAEA and other external organizations.
- Second GIF Symposium: In November 2012, the second GIF Symposium was held in San Diego, USA. The first open session, as part of an ANS meeting, provided wide range of non-GIF audience with the update of our activities. At the second closed session, we had an active exchange of ideas regarding future strategic planning among GIF participants. Through the symposium, the importance of sharing information on GIF's activities with more people has been recognized.

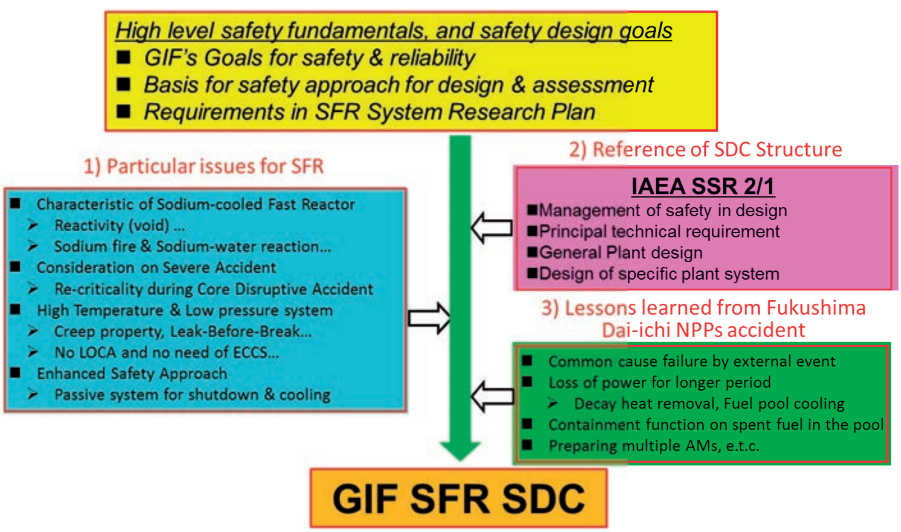


FIG. 4. Basic scheme outlining the SDC.

FIG. 5. GIF Symposium in San Diego, USA (November 2012).

— Changes in circumstances surrounding GIF.
- The long term security of natural resources and energy is a subject of continuing importance for humanity, considering the future increase in energy needs. In order to sustain global energy supplies, dependence on nuclear energy will steadily increase. Even after the accident at TEPCO Fukushima Daiichi NPPs two years ago, which reminded the world of the importance of nuclear safety, nuclear energy policies and development plans almost remain unchanged in the majority of countries although some countries have changed direction of nuclear strategy.
- In China, India and the Russian Federation, advanced reactor construction projects are being advanced and more activities on the Gen-IV R&D are expected to be carried out.

5. FUTURE DIRECTION OF GIF

GIF has completed the first decade last year since the conclusion of the GIF Charter and the formulation of the technology roadmap. As we have established a framework of international cooperation of Gen-IV R&D, it can be said that we have almost accomplished our objective of the first stage.

In light of these circumstances, we set up a strategic planning TF under the PG in order to plan activities in the next decade aiming at further development of GIF activities. The TF is conducting a review of the present status and recommendations on three subjects: updating the technology roadmap; strengthening R&D collaboration within GIF; and strengthening ties with other international organizations.

The status of progress in R&D will be reflected in the new technology roadmap because differences in the maturity of development level has been recognized in the last ten years since the start of GIF. For example, the SFR system is in the demonstration phase with several demonstration projects, on the other hand, the MSR is still in the validation phase to be confirmed as a Gen-IV system. In the past decade, the GIF served as an international community for establishing the viability and performance of each Gen-IV system. In the new decade, however, we need to consider developing international criteria or guidelines on safety and ISI in the R&D on the systems in the demonstration phase. Meanwhile, for promoting collaborative activities, we have started research and examination on the possibilities for the collaborative use of facilities in Member States. We also intend to develop human resources engaged in R&D in the future.

6. CONCLUSION

I hope that in this FR13, active exchange of information will be made on fast reactor development, GIF's main target, and that R&D of the Gen-IV systems with respect to their demonstration phases will be steadily promoted on a global basis.

NATIONAL AND INTERNATIONAL FAST REACTORS PROGRAMMES

(Plenary Session)

Chairpersons

V. PERSHUKOV
Russian Federation

P.R. VASUDEVA RAO
India

C. BEHAR
France

FAST REACTOR DEVELOPMENT STRATEGY IN CHINA

DONGHUI ZHANG, YUANYU WU, HONG YU
China Institute of Atomic Energy,
Bejing, China

Abstract

As one of the largest developing countries, China needs a reliable energy supplement. At the same time, China should improve the energy structure to decrease CO_2 emissions. Nuclear and renewable energies are the main solutions to these issues. According to the research results, the nuclear capacity should increase to 400 GW(e) up to 2050. Fast reactors must be developed considering the limitation of uranium resources. In order to deploy fast reactor technology, the 'experimental reactor, demonstration reactor and commercial reactor' strategy has been suggested. China has finished the construction of the China Experimental Fast Reactor (CEFR) and gained necessary experience about fast reactors. The China Institute of Atomic Energy (CIAE) has begun to design the CFR-600, a 600 MW(e) demonstration fast reactor. This reactor will be put into operation before 2025. After that, a larger commercial reactor will be constructed. Besides fast reactors, all of other key sectors of fuel cycle will be developed at the same time such as reprocessing, fast reactor fuel, etc. There are two main tasks of fast reactors, one of which is to raise the utility ratio of uranium, and the other one is to transmute the long life waste of light water reactors. The fast reactor will be designed as a breeder and burner, respectively.

1. BACKGROUND

China is one of the largest developing countries in the world. At the same time, China is the world's largest factory, which means China needs to consume a large amount of energy to produce goods to support the global market. As one of the basic essentials for production, energy is always a key issue in China. From the very beginning, the Chinese government pays much attention to the energy supply and keeps the energy self-support ratio higher than 90%. But the problem is that the structure of energy supply is not good; coal occupies 80% of primary energy. The total energy consumption is about 3.2 billion tonnes of coal in recent years. This induces two main issues, one of which is the supply unsustainability because coal reserves are limited, and the other one is the greenhouse effect caused by CO_2 emissions. In order to solve these issues, clean and renewable energy development is encouraged, including hydropower, solar energy, wind power, nuclear power, etc. The total capacity of clean energy will rise to about 15% of primary energy [1].

The development of nuclear energy is very rapid in China. There are 11 units which have been put into operation and 24 units are under construction. After the Fukushima nuclear accident, the Chinese government took very strict actions in the safety inspect of nuclear power plants and began to pay more attention to the safety of nuclear power. All construction projects were suspended, and the experts' team checked all the nuclear power plants and all other nuclear facilities. The results showed that China's nuclear power plants are generally in good condition, with only a few inadequacies, such as movable diesel generators, anti-tsunami measures, etc. The regulatory body asked the operators to work on plant weaknesses to meet the new safety requirements after the Fukushima accident. In order to raise the safety level of the nuclear industry, the government issued two plans, one is the "Nuclear safety development middle–long time plan", and the other is the "Nuclear power development middle–long time plan".

After careful assessment, the national energy authority and national nuclear safety authority permitted some projects to restart recently. The nuclear power capacity is estimated to be about 70 GW(e) up to 2020 and the total nuclear capacity may be higher than 400 GW(e) up to 2050. It is very difficult to complete this task because of the limitation of uranium resources if only PWR plants are built. Fast breeder reactors and the fuel cycle must be developed under these circumstances.

2. FAST REACTOR DEVELOPMENT STRATEGY IN CHINA

2.1. Scenario of future nuclear power

Studies show that the Chinese population will be 1.43 billion and the primary energy consumption will be 3.5 tsce/year/person in 2050 [2]. Total energy consumption will be 5 billion tsce. The total electricity capacity will be 2.5 billion kW. Table 1 shows the increase of nuclear power.

In order to accomplish the goal of nuclear power development, fast reactors must be developed at the same time. It is believed that fast reactor technology will reach a commercial level by 2030. There are two scenarios for fast reactor development. In the first scenario, there is a sufficient uranium resource. The main task of fast reactors is to transmute minor actinides of PWR spent fuel. In this case, the capacity of fast reactors is about 70 GW(e), which is 20% of PWR capacity. In the second scenario, there is an insufficient uranium resource. The main task of fast reactors is to breed. Then, fast reactor capacity will be more than 200 GW(e), which is 50% of nuclear power at least.

TABLE 1. THE INCREASE OF TOTAL CAPACITY AND NUCLEAR POWER (BILLION kW)

	2010	2020	2030	2050
Total capacity	0.95	1.5	2.0	2.5
Nuclear	0.01	0.07	0.2	0.4
Capacity ratio of nuclear	1.05%	4.6%	10%	16%
Electricity generation ratio of nuclear		7%	15%	24%

It is necessary to mention the fuel cycle because the fuel for fast reactors is made from the spent fuel of PWRs. China adopts a closed fuel cycle approach to sustain the development of fission energy (Fig. 1). The industrial system of mining, refining, enrichment, UO_2 fuel fabrication and PWR power plants has been established. For reprocessing, a pilot reprocessing plant with the capability of 50–80 t/y has been put into use and an industrial-size reprocessing plant has been identified. There are two main options for fast reactor fuels in China: mixed oxide fuel (MOX) and U-Pu-Zr alloy. A laboratory size MOX fuel line is under construction and an industrial MOX fuel plant is under discussion. Some studies about metal fuel have been done and more research work will be carried out in the future.

2.2. Fast reactor development strategy

A three-step strategy is adopted to develop fast reactor technology in China. The first step is to develop an experimental fast reactor, which has been realized. The experimental reactor (CEFR) was put into operation in 2011. A demonstration fast reactor is the second step and the third step is to develop commercial fast reactors. For the next steps, the power capacity of the demonstration and commercial fast reactors have been chosen as 600 MW(e) and 1000 MW(e), respectively. An inland site located in Fujian province was selected as the prior site for the CFR600. The plan is to finish the construction of the CFR600 before 2025 and to realize fast reactor commercialization around 2030.

China initiated fast reactor research from the middle of the 1960s. The research work mainly focused on the neutron physics, thermohydraulics, fuel, material, sodium technologies, safety, etc. After 20 years of research, China mastered most of the basic knowledge about fast reactors and from 1986, China started to study the design principles in order to design an experimental

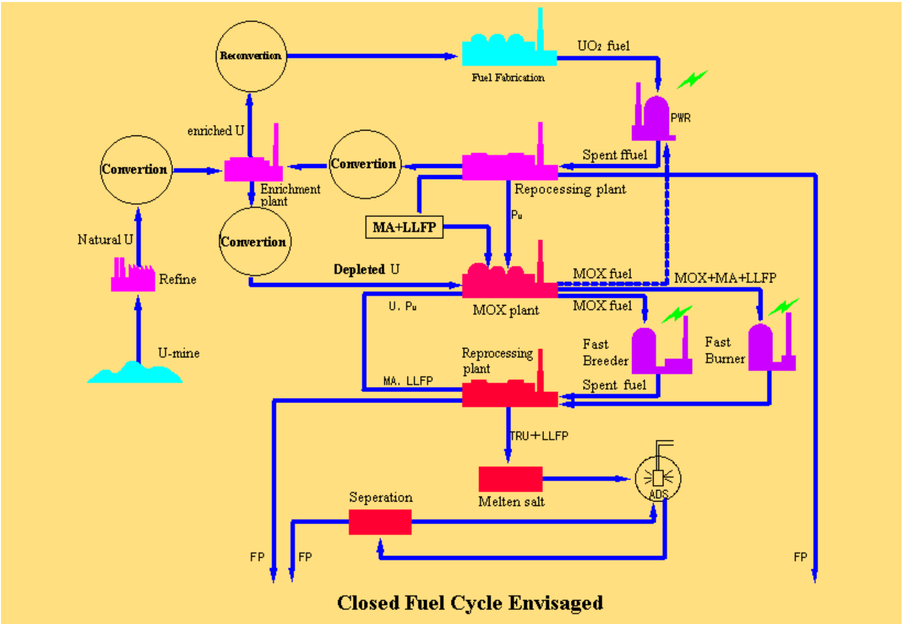


FIG. 1. Fuel cycle in China.

reactor. Before the end of 1995, the China Experimental Fast Reactor project had been identified.

The CEFR is a sodium cooled 65 MW(th) experimental fast reactor. In order to gain more experience for the development of a demonstration fast reactor, CEFR has adopted a pool type design with electricity generation function. The main systems and parameters are very close to large fast reactors. Table 2 gives the main parameters of the CEFR and Fig. 2 shows the main heat transfer system.

3. THE DEMONSTRATION FAST REACTOR

The principles for the demonstration reactor design are:

— It should be large enough to make industrial verification;
— The technology should be advanced enough to meet the requirements for nuclear power plants several decades later;
— The technology and economy risks should be low enough;
— The whole fuel cycle should be considered.

TABLE 2. MAIN PARAMETERS OF CEFR

Parameter	Unit	Value
Thermal power	MW	65
Electric power (net)	MW	20
Reactor core		
Height	cm	45.0
Diameter equivalent	cm	60.0
Fuel		MOX (first loading is UO_2)
Linear power (max.)	W/cm	430
Neutron flux	$n/cm^2\cdot s$	3.7×10^{15}
Burnup	MWd/t	60000
Inlet/outlet temp. of the core	°C	360/530
Diameter of main vessel (outside)	m	8.010
Design life	A	30
Primary circuit		
Number of loops		2
Mass of sodium	t	260
Flow rate (total)	t/h	1328.4
Number of IHX per loop		2
Secondary circuit		
Number of loop		2
Quantity of sodium	t	48.2
Flow rate	t/h	986.4

TABLE 2. MAIN PARAMETERS OF CEFR (cont.)

Parameter	Unit	Value
Tertiary circuit		
Steam temperature	°C	480
Steam pressure	MPa	14
Flow rate	t/h	96.2

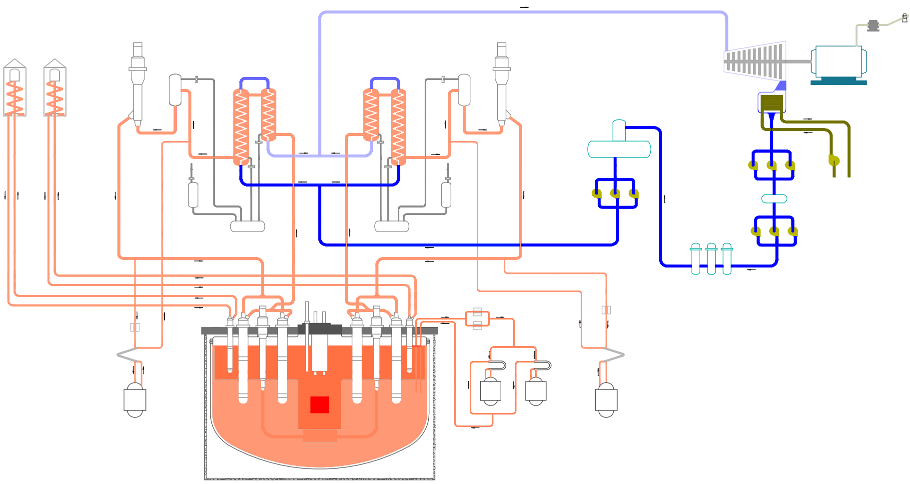


FIG. 2. Main heat transfer system of the CEFR.

The demo reactor CFR600 is a sodium cooled, pool-type reactor which is very similar to the CEFR. The electricity capacity is 600 MW. This reactor is a middle-size reactor, which could be designed as breeder and burner in one reactor core. The main design features of the CFR600 follow the requirements for the fourth generation nuclear technology: that it is sustainable, safe and reliable. The main input design parameters are given as follows:

— ~1500 MW(th)，600 MW(e);
— MOX fuel;
— Breeding ratio: 1.2;
— Sodium as coolant;
— Na-Na-H_2O loops with 2 primary and secondary loops;

— One turbine;
— Negative feedback;
— Confinement;
— CDF<10^{-6}, the probability of a large-scale release of radioactivity<10^{-8};
— Design Life: 60 a.

The schematic diagram of the CFR600 is given in Fig. 3.

The design of the CFR600 was initiated in August 2012. The concept design will be finished this year and the primary design will be completed in 2015. At the same time, research work has been carried out on some key equipment and technology.

4. CONCLUSION

China devotes itself to the peaceful use of nuclear to meet the growing energy demand. An adequate number of nuclear power plants could provide clean energy with low risk. This is very important for ensuring the positive outcomes of modern industrial civilization with as little damage to the environment as possible.

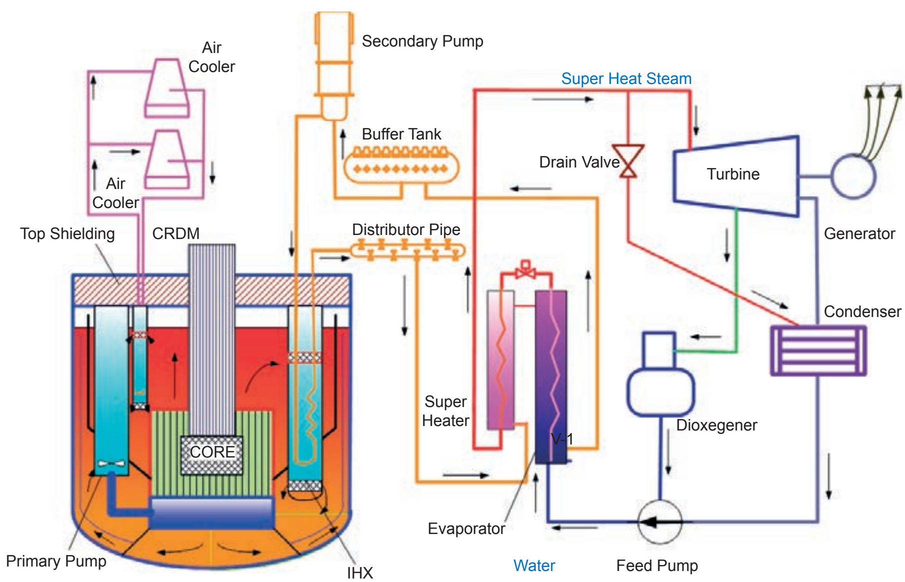


FIG. 3. The schematic diagram of the CFR600.

The fast reactor is a promising technology to ensure the sustainable development of nuclear energy, which can produce new fuel from depleted uranium and burn long life radioactive waste at the same time. Sodium cooled fast reactor technology is one of the six recommended Generation IV technologies with inherent safety features. It is expected that fast reactors will provide enough clean power to people for the long term future.

REFERENCES

[1] RESEARCH TEAM OF MEDIUM AND LONG TERM DEVELOPMENT STRATEGY OF CHINA'S ENERGY, Study on Medium- and Long-term Development Strategy of China's Energy. Electricity, Oil and Gas, Nuclear, Environment Volume. Science Press (2011), China's Medium and Long-term Energy Development Strategy, Science Press, Beijing (2011)

[2] RESEARCH TEAM OF MEDIUM AND LONG TERM DEVELOPMENT STRATEGY OF CHINA'S ENERGY, Study on Medium- and Long-term Development Strategy of China's Energy, Comprehensive Volume, Science Press, Beijing (2011).

THE FRENCH R&D PROGRAMME ON SFRs AND THE ASTRID PROTOTYPE*

C. BEHAR
Commissariat à l'énergie atomique et aux énergies alternatives,
Saclay, France

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

A PERSPECTIVE ON THE INDIAN PROGRAMME ON FAST REACTORS AND ASSOCIATED FUEL CYCLES

P.R. VASUDEVA RAO, P. CHELLAPANDI, G. SRINIVASAN,
K.K. RAJAN, R. NATARAJAN, P.V. KUMAR, T. JAYAKUMAR,
S.C. CHETAL
Indira Gandhi Centre for Atomic Research,
Kalpakkam, Tamil Nadu,
India

Abstract

Fast reactors are expected to make a significant contribution to nuclear energy production in India in the coming decades. In line with this approach, India is pursuing an intense and comprehensive programme of development of fast reactors and associated fuel cycles. The Fast Breeder Test Reactor (FBTR) has been successful as a tool for irradiation experiments on fuels and advanced materials, qualification of instrumentation and equipment and as a resource for training of personnel for operation and maintenance. These apart, it has provided a vast operating experience of sodium systems, key components and sodium technology. Construction of the 500 MW(e) Prototype Fast Breeder Reactor (PFBR) is nearing completion. R&D and qualification of reactor components has been almost completed and the commissioning activities have started. A test subassembly of 37 pins of PFBR fuel (MOX) has been irradiated in FBTR to a peak burnup of 112 GW·d/t. Manufacturing of the fuel and the subassemblies for the reactor is in progress. Towards establishing the next fast reactors in the series, several advanced design features to achieve improved economy and enhanced safety are under consideration. These design features, for example, would result in extended plant life (60 years), reduction in specific steel requirements (up to 25%), increased burnup, enhanced diversity in shutdown system, decay heat removal systems enhanced in-service inspection capabilities, etc. The long term approach is to exploit the high breeding potential of the metal alloy fuelled fast breeder reactors to enhance the rate of expansion of the fast reactor capacity. Towards establishing the metal fuelled reactors, R&D in a variety of domains has been initiated, which includes the fabrication and irradiation of sodium bonded and mechanically bonded metal alloy test pins in FBTR. Simultaneous with reactor development, the activities of establishing the associated fuel cycles have also been pursued. The mixed carbide fuel of FBTR irradiated up to 150 GW·d/t has been reprocessed and the closure of the fuel cycle of FBTR demonstrated. The development of pyroprocessing technology for metallic fuels is being undertaken on a lab scale for understanding the science behind the process steps and on an engineering scale to develop the equipment and processes that can be subsequently translated to plant scale. India is also addressing the gamut of other technologies associated with the development of fast reactors and fuel cycles including materials development and safety studies, inspection technologies, components testing, instrumentation development, etc.

1. INTRODUCTION

Currently, the share of nuclear power in India is less than 3% of the electricity production. However, with the maturity of nuclear technology, growth in industrial capabilities and the large demand for energy, nuclear energy in India is expected to grow significantly in the coming decades. With limited uranium resources available indigenously, there is a large emphasis on the development of fast reactors with closed fuel cycle to exploit the uranium resources effectively. Accordingly, fast reactors are expected to contribute a major share of nuclear energy production in the coming decades.

India has successfully operated the Fast Breeder Test Reactor (FBTR) for nearly three decades. The operation of this reactor, combined with a comprehensive R&D programme in all domains relevant to fast reactors and the associated fuel cycles, has generated the expertise and knowledge base necessary for launching a large fast reactor programme. As a first step, a 500 MW(e) Prototype Fast Breeder Reactor (PFBR) is under construction at Kalpakkam and a dedicated co-located Fast Reactor Fuel Cycle Facility (FRFCF) is planned to cater for this reactor. As the long term option, India is pursuing the development of metal fuelled FBRs to exploit their breeding potential towards enhanced growth of the fast reactor capacity. Developmental actions in this regard are in progress at IGCAR and BARC. This paper provides an overview of the status of the programme in India on fast reactors and associated fuel cycles.

2. FAST BREEDER TEST REACTOR

The FBTR, which went critical in October 1985, has been working satisfactorily. It started with a small core of 23 fuel subassemblies of a unique high Pu MK-I carbide fuel ($U_{0.3}Pu_{0.7}C$). The core has since been progressively expanded by inducting MK-II fuel ($U_{0.45}Pu_{0.55}C$) surrounding the MK-I fuel. Based on the post-irradiation examination (PIE) of the MK-I fuel at different burnup levels, it was found feasible to extend the burnup limit of MK-I fuel to as high as 150 GW·d/t. The expansion of the core, coupled with raising of the linear heat rating of the MK-I fuel based on PIE, have enabled raising the core power rating to 22.5 MW(t). More than 1200 MK-I fuel pins have seen the burnup of 155 GW·d/t. There has been only one fuel failure so far, at 146 GW·d/t. The failed fuel was easily identified in just two fuel handling campaigns using a combination of different techniques, i.e. DND count contrast ratio, $^{85}Kr/^{87}Kr$ ratio in cover gas and flux tilting.

The MOX fuel test subassembly designed for the PFBR was irradiated in the FBTR to a peak burnup of 112 GW·d/t before discharge. PIE of the fuel has given confidence in successfully deploying it in the PFBR. Structural and shielding materials of interest to fast reactors are undergoing irradiation tests. The trial production of Sr^{89}, an isotope used as a palliative in bone cancer, was demonstrated by irradiating yttria and subsequent separation in hot cells. The current major mission of the FBTR is towards the development of metallic fuels, which are foreseen as the potential fuels for the future fast reactor programme of India. As a forerunner to validate the sodium bonded metallic pin fabrication process, U-Zr pins are already undergoing irradiation tests. It is planned to irradiate U-19Pu-6Zr pins in the coming campaigns.

As a part of ageing management, several refurbishments have been carried out, such as replacement of steam generator rupture discs, fire water system, main boiler feed pumps, etc. Seismic reevaluation of the plant to current standards has been completed. While some of the modifications will be completed by June 2013, all the retrofits are scheduled to be completed by December 2014. Residual life assessment of the reactor components has been carried out. Creep-fatigue damage of high temperature components was calculated from plant operating history and showed negligible damage. Loss of ductility of the grid plate due to neutron dose was found to limit the residual life of the reactor. However, it has been confirmed that the grid plate has so far seen only ~1 dpa, as compared to the acceptable value of 4 dpa for meeting the residual ductility criterion assuring a long residual life for the reactor. The post-Fukishima safety evaluation has been completed. The plant is seen to be safe against extended blackout. Retro-fits are being planned for protection against flooding.

3. PFBR

3.1. Project status

All the major and heavy items of reactor equipment have been erected. In particular, the equipment below the roof slab inside the main vessel has been erected. Erection of long reactor components, i.e. intermediate heat exchangers (IHX), air heat exchangers (AHX) and the inclined fuel transfer machine (IFTM) are in progress. The steam generators have been erected (Fig. 1). Absorber drive mechanisms (CSRDM and DSRDM) will be erected after erection of the control plug. Turbogenerator equipment are in an advanced stage of erection. Commissioning of support systems is progressing in parallel. Erection of sodium piping is nearing completion. Main plant I&C works have been completed. The training simulator has been fully installed. Software has been loaded and training

FIG. 1. Steam generators erected at the PFBR.

is in progress. In parallel, licensing of the operators by the regulatory agency is in progress. The O&M personnel of the PFBR have been trained in the FBTR. As of the end January 2013, the project had achieved a physical progress of 93%.

3.2. Component development and performance qualification

Considering the fact that the design and manufacture of many components of the PFBR have been undertaken indigenously and that many of the components were manufactured for the first time by Indian industry, testing of critical components at full scale and in simulated reactor experiments was undertaken to generate confidence in the performance of the components. The control and safety rod drive mechanism (CSRDM) and the diverse safety rod drive mechanism (DSRDM) were qualified for 14 years and 10 years of reactor operation, respectively. The CSRDM and DSRDM required for the first criticality of the PFBR have been delivered. The transfer arm and IFTM are deployed in the PFBR for handling of the subassemblies. The critical in-sodium components of the IFTM, namely primary ramp and primary tilting mechanism (Fig. 2), were tested in sodium and have been installed in the reactor. In-sodium testing of the transfer arm is in progress. Several other components such as electromagnetic pumps and eddy current flowmeters, have been developed indigenously and delivered to the project site after performance testing.

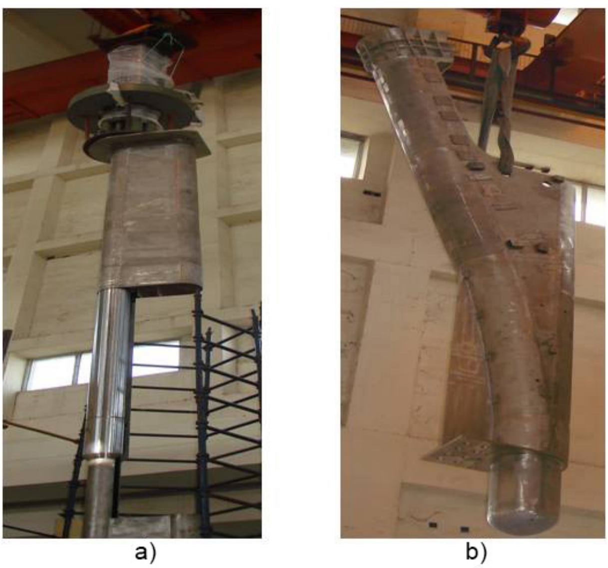


FIG. 2. (a) Assembly of transfer arm; (b) primary tilting mechanism.

A number of sensors and probes for use in the reactor have been designed and manufactured indigenously. The level probes for measuring the level of sodium in various tanks as well as the main vessel have been delivered to the PFBR for testing and calibration. An under sodium ultrasonic scanner is being developed for viewing the protrusion, growth and bowing of the fuel subassemblies in the reactor. A prototype scanner has been developed and in-sodium testing is in progress.

3.3. In-service inspection (ISI) of PFBR components

For the ISI of main vessel and safety vessel (SV), a remote-controlled 4-wheeled robotic vehicle with two wheels resting on each vessel in the interspace has been designed. This robotic vehicle is designed to carry visual and ultrasonic non-destructive evaluation (NDE) modules. For the ISI of shell weld of core support structure (CSS), which is welded to the main vessel skirt at the bottom, a novel ultrasonic NDE methodology has been developed. This methodology has been successfully demonstrated on a main vessel sector mock-up assembly and detection of defects down to 20% wall thickness (~6 mm) on both sides of the plate has been reliably achieved. Another robotic device has been designed for carrying out ultrasonic ISI of dissimilar welds above the anti-convection barrier and in-situ welds below the anti-convection barrier. In order to carry out periodic ISI of steam generator tubes made of modified 9Cr-1Mo steel, comprehensive remote field eddy current (RFEC) technology comprising of instrument, flexible probe, methodology and robotic device has been developed. Using this technology, detection of wall thickness loss down to 20% has been successfully demonstrated, despite the presence of sodium in defects. Wavelet transform based signal processing method has been developed to suppress RFEC signals from expansion bend regions.

3.4. Performance evaluation of the PFBR MOX test fuel assembly

PIE was carried out on the 37-pin test sub-assembly of PFBR MOX fuel having alloy D9 fuel cladding and wrapper that was irradiated in FBTR up to a burnup of 112 GW·d/t at a peak linear heat rating of 450 W/cm. Both non-destructive and destructive examinations were carried out. Dimensional variations in wrapper and fuel pins were found to be well within the margins provided for in the design. Void swelling ($\Delta V/V$) of the fuel cladding tube irradiated to 50–60 dpa at ~420–500ºC is about 1.2–2%, which is much less than the value of 3–5% for CW SS 316 stainles steel at a similar damage level. Radiography and gamma scanning of the fuel pins showed that the maximum fuel stack length increase is about 3.2 mm, indicating that the swelling of fuel is as expected. Cladding tube wastage, the central hole diameter of fuel pellet and pellet-fuel-cladding tube gap variations are all within the allowances provided in the design. The tensile properties of the alloy D9 cladding tube and wrapper specimens at displacement damages of 40–60 dpa showed an increase in strength (hardening) with a reduction in ductility. The results revealed that the PFBR test subassembly has performed well without any indications of failure and that sufficient margins are available on cladding tube performance. The data generated on the PFBR fuel and cladding material have enhanced our confidence in the performance of fuel subassemblies in the PFBR for achieving the design target burnup.

3.5. Safety studies

Following the accident at Fukushima, several safety analyses and experiments have been taken up. The analyses include assessment of safety margins of reactor components and safety related civil structures under earthquakes beyond SSE, design basis tsunami indicating protections for higher flood level, sequential leakage of main and safety vessels, simultaneous double ended guillotine rupture of two primary pipes, multiple failure of safety grade decay heat removal heat exchangers (SGDHRs), hydrogen generation in reactor containment building and need for hydrogen recombinators. Thermal analysis of the spent fuel storage bay under a prolonged station blackout condition has been carried out for different decay power load conditions of the water storage pool. From the analysis, it is concluded that the water loss can be very well made-up by adopting a strategy of valving in the fire water system. This exercise forms the basis for planning the plant emergency operation procedures.

With regard to addressing the possible sodium leak from sodium circuits in the steam generator buildings, a passive leak collection arrangement, including leak collection tray (LCT), is incorporated in the design. The evaluation of LCT performance was carried out in the Sodium Fire Experimental Facility, with 40 kg sodium at 550ºC leaked into the LCT at the rate of 0.5 kg/s. Based on the tests, further improvement and optimization of LCT design is envisaged in the next stage with full scale demonstration of the total LCT system.

SOdium–CAble (SOCA), an integrated experimental facility, has been commissioned to investigate the possible scenarios of sodium fire followed by cable fire (i.e. secondary fire) on the top shield platform, consequent to CDA. The tests focus on the structural integrity of the SGDHR piping passing through the platform and on the structural integrity of the SGDHR piping above the top shield.

FIG. 3. Grid plate melt through tests.

To confirm the availability of natural convection path for the post accident heat removal purpose, i.e. perforation in the grid plate by the molten fuel and structural materials consequent to CDA, a series of tests were performed with Woods metal in water, simulating the hydrodynamic and thermal characteristics of molten UO_2 (Fig. 3). The melt through was observed to happen in the centre of the grid plate. The data collected through digital imaging and processing with high speed photography form important inputs for the validation of a computer code, which is under development.

Heat transfer experiments conducted with the PFBR model steam generator have established the area margin available. The effectiveness of the near passive PFBR decay heat removal systems was successfully demonstrated by experiments with sodium in the SADHANA facility.

4. DEVELOPMENT OF FUTURE FBRS

4.1. Design concept

Design activities related to two more 500 MW(e) fast breeder reactors (FBR 1&2) with improved economy and enhanced safety, are in progress. These reactors will be constructed at Kalpakkam. In view of the fact that fast reactors are capital cost intensive projects, the capital cost needs to be brought down significantly. Use of twin unit concept for sharing many non-safety related systems and services, use of advanced shielding materials such as ferro-boron, SS 304 LN in place of SS 316 LN for cold pool components and ferritic steels for sodium piping, 3 steam generator modules per loop with an increased tube length of 30 m (the PFBR has 4 modules per loop with 23 m length), enhancing the design life from 40 years to 60 years and enhanced burnup to 200 GW·d/t in stages have indicated sizeable capital cost and tariff reduction.

Calibrated improvements are planned in the design of rector assembly components along with simplification of the fuel handling scheme, taking advantage of the twin unit concept. The improvements conceived in the reactor assembly shown in Fig. 4 are: (1) reduction of main vessel diameter, (2) dome shaped roof slab with conical support skirt under compression, (3) thick plate concept for the rotatable plugs, (4) welded grid plate with reduced number of sleeves, reduced diameter of intermediate shell and reduced height, (5) increased number of primary pipes, (6) inner vessel with single radius torus welded with the grid plate, (7) integrated liner and safety vessel with thermal insulation arrangement and (8) optimization of main vessel, inner vessel and safety vessel thicknesses. The IFTM has been eliminated with an addition of two straight pull machines. These improvements are expected to bring about cost benefits with enhanced safety.

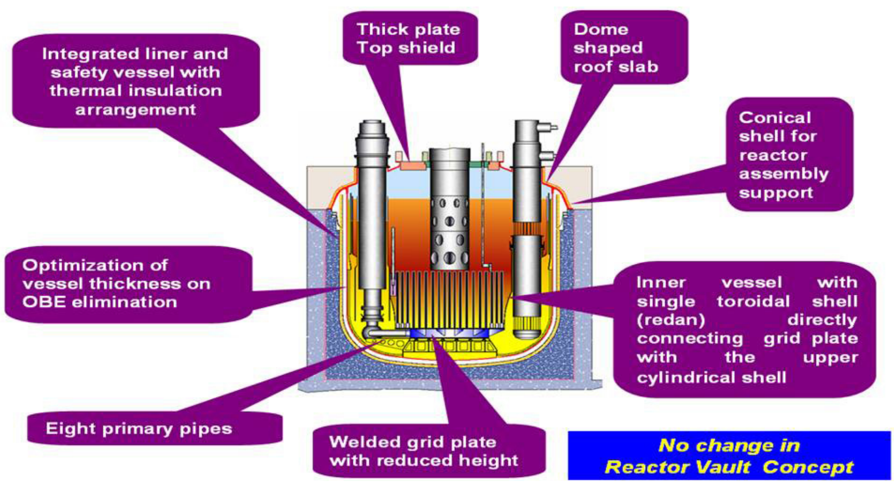


FIG. 4. Improvements proposed in the FBR 1&2 reactor assembly.

4.2. Technology development

Technology development exercises have been completed for those components (Fig. 5) where the design challenges are significant, i.e. tri-junction forging for dome shaped roof slab, large diameter bearing, thick plate welding for rotatable plugs, welded grid plate, inner vessel with redan of large single torus and 30 m long tubes of steam generator.

4.3. Development of advanced clad materials for sodium cooled fast reactors

Oxide dispersion strengthened (ODS) ferritic martensitic steel Fe-0.1C-9Cr-2W-0.2Ti-0.34Y_2O_3 (wt%) has been developed for SFR fuel cladding tubes. A complex powder metallurgy route followed by hot and cold mechanical processing was adopted to produce the cladding tube. Cladding tubes with 6.6 mm outer diameter, 0.45 mm thickness and up to 4500 mm length have been successfully produced. Long term creep rupture strength of the developed 9Cr-ODS ferritic martensitic steel fuel cladding tube was found to be comparable to that of the alloy D9 cladding tube (Fig. 6).

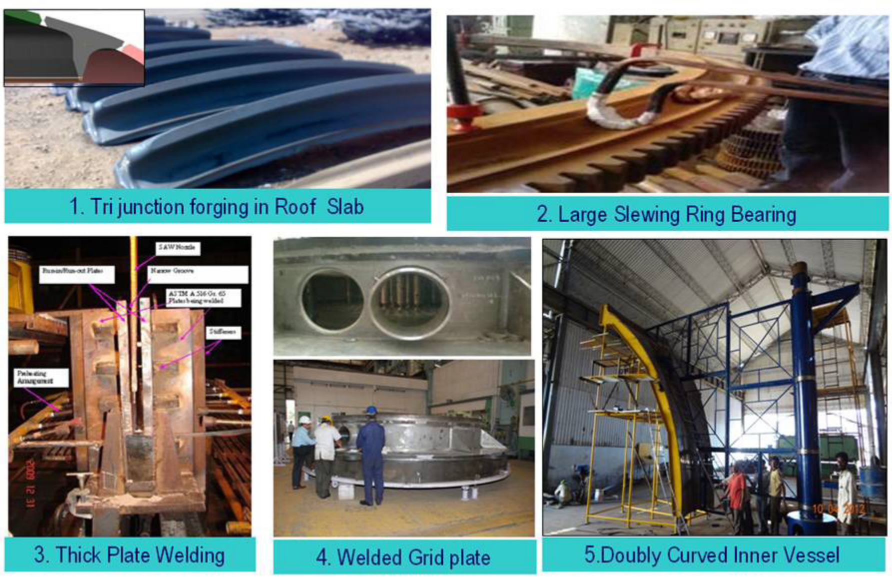


FIG. 5. Technology development for design confirmation.

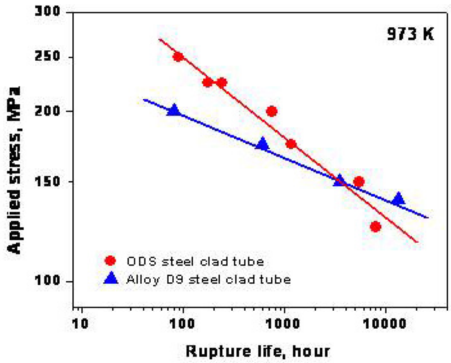


FIG. 6. Comparison of creep rupture strength of ODS and alloy D9 steel cladding tubes.

In an effort to optimize the alloy composition around the nominal alloy D9 levels and to identify an improved version of alloy D9 having higher resistance to both void swelling and creep, a series of laboratory heats were conducted with varying compositions: 0.025–0.04 wt% phosphorous, 0.75–0.95 wt% silicon and 0.16–0.30 wt% of titanium. Based on the results from extensive creep tests and ion irradiation studies for void swelling, an optimized alloy D9 alloy having 0.25 wt% Ti, with 0.75 wt% Si and 0.04 wt% P, designated as Indian Fast Reactor Advanced Cladding-1 (IFAC-1), has been developed for fuel cladding tubes in SFR. The alloy IFAC-1, with optimum composition of minor elements is expected to allow safe operation up to a damage level of ~150 dpa.

5. FUEL CYCLE

5.1. Reprocessing of FBTR fuel

The reprocessing of U, Pu mixed carbide fuel discharged from the FBTR has been continued in the CORAL facility and the campaigns have provided valuable inputs towards the design of the Fast Reactor Fuel Cycle Facility. Mixed carbide fuel discharged up to a burnup of 150 GW·d/t and a cooling time as short as two years, has been reprocessed successfully. The Pu recovered from the reprocessing campaigns has been used in the fabrication of fresh fuel for the FBTR and one subassembly fabricated using the recovered Pu has already been loaded in FBTR, thus demonstrating the closure of the fuel cycle. Improvements introduced in the reprocessing flowsheet have resulted in better product purity and lower waste volumes. For the reprocessing of FBTR fuel on a regular basis, a Demonstration

Fast Reactor Fuel Reprocessing Plant (DFRP) has been constructed based on inputs from CORAL. This plant is in an advanced stage of completion. This plant is designed to take up the reprocessing of the initial discharges of MOX fuel from the PFBR.

5.2. Fuel cycle for the PFBR

A Fast Reactor Fuel Cycle Facility is planned to be constructed to close the fuel cycle of the PFBR. This facility is located close to the PFBR at the Kalpakkam complex and thus would avoid transport of fresh and irradiated fuel in the public domain and also provide savings in transport costs. The pre-project activities for the construction of the facility have been completed and the construction is expected to start later during the year. When completed, this facility would have all the plants required for the closing fuel cycle of the PFBR, including the fuel fabrication plant, fuel reprocessing plant, fuel assembly plant and waste management plant.

Manufacture of the radial blanket pins of the PFBR has been completed. Fabrication of the MOX fuel is under way at the Advanced Fuel Fabrication Facility at Tarapur. The work on the constitution of the subassemblies of fuel and blanket has already commenced at a special facility constructed for this purpose.

5.3. Sol–gel fuel fabrication

The sol–gel process is a fuel fabrication route with good potential for fast reactor fuel fabrication and also fabrication fuels containing minor actinides. In collaboration with BARC, IGCAR has established a facility to fabricate fuel pins containing uranium, plutonium mixed oxide microspheres. In this facility, fuel pins containing a mixture of $(U,Pu)O_2$ microspheres (Pu content: 53%; size 780 + 70 microns) and UO_2 microspheres (115 + 10 microns) have been fabricated. As the thermal conductivity of the Pu-rich mixed oxide is low, the pin was filled with helium at around 6 bar pressure. The pins have been qualified by non-destructive examination and are assembled into a test capsule along with pins containing pellets of MOX with an average Pu content similar to that of the microspheres, for test irradiation in FBTR.

5.4. Metal fuelled FBRs and their fuel cycle

Metal fuelled FBRs are being pursued as a long term option to exploit their potential for breeding and enable an enhanced growth in capacity for power production through fast reactors. India has initiated a programme for the development of mechanically bonded (U-Pu alloy with Zr liner) and sodium bonded (U-Pu-Zr) fuels. U-Pu-Zr alloy slugs with 6% Zr were used to fabricate sodium bonded pins. Clad material was modified 9Cr-1Mo. Trial runs with U-Zr pins have been completed and the pins are undergoing irradiation in the FBTR. Irradiation of U-Pu-Zr pins is expected to be started in the next campaign of the FBTR. A photograph of the facility for fabrication of sodium bonded fuel pins is shown in Fig. 7.

Development of pyroprocessing technology is being pursued as it is the primary choice for the metal fuel cycle. A combination of laboratory scale experiments with Pu alloys and engineering scale U alloys is being pursued. A view of the engineering scale facility is shown in Fig. 8. In this facility, a trial run on chopping of sodium bonded U-Zr fuel pins, molten salt electrorefining of U-Zr and consolidation of the deposit have been carried out on a kilogram scale.

FIG. 7. View of the facility for fabrication of sodium bonded fuel pins.

FIG. 8. View of the electrorefiner and other equipment inside inert atmosphere glovebox.

An ambient temperature electrorefiner test facility has been set up to develop remote operation concepts for the electrorefiner on the 10 kg scale. In this set up, experiments will be carried out using copper sulphate solutions. A High Temperature ElectroRefiner (HTER) is planned to be set up to demonstrate the electrorefining operation on the 10 kg scale.

6. CONCLUSION

India is pursuing developmental activities on all facets of the fast reactor programme, with the aim of establishing a large nuclear capacity based on fast reactors with closed fuel cycle in the coming decades. The successful operation of the FBTR and the reprocessing of FBTR fuel have provided the confidence for launching the fast reactor programme in the commercial domain. Commissioning of various systems of PFBR, a techno-economic demonstrator and also a forerunner in the series of SFRs, has commenced. Several advanced design features to achieve improved economy and enhanced safety are being considered for the reactors planned beyond the PFBR. Introduction of metallic fuelled reactors with emphasis on breeding and enhanced safey is planned in the long term. Towards this, systematic and comprehensive R&D activities are under way. Many other aspects such as knowledge management, human resource development, academic collaborations and partnership with industry are being addressed to sustain and enhance the programme and ensure its success.

DELIBERATION OF POST-3.11 FAST REACTOR R&D STRATEGY IN JAPAN

S. KONDO
Japan Atomic Energy Commission,
Tokyo, Japan

Abstract

The severe accident on 11 March 2011 at the Fukuhsima Daiichi nuclear power plant in Japan has changed the public perception towards nuclear power and it is now predicted that nuclear power will decrease its contribution to energy supply in the long term. Taking this prediction into consideration, Japan is in the process of deliberating the future R&D strategy for the SFR and its fuel cycle technology. Currently, the following are under discussion as major R&D objectives for the SFR and its fuel cycles technology: (1) to accomplish the full power operation of MONJU, improving its safety feature in accordance with new safety standards to be set by the NRA; (2) to explore innovative fast reactor technology for reducing the amount and toxic level of radioactive waste, (3) to explore innovative technologies for strengthening the safety of SNRs and (4) to support innovation in nuclear energy technology, safety, waste management, non-proliferation and security relevant to fast reactors and their fuel cycle technology. As these are major objectives of R&D for a sustainable nuclear energy system, Japan will carry out activities to achieve these objectives in close cooperation with the international community.

1. INTRODUCTION

The Great East Japan earthquake and the resulting tsunamis struck people and facilities, including nuclear power plants, located on the Pacific coast of Japan on 11 March 2011. This caused an unprecedented severe accident at the Fukushima Daiichi nuclear power plant operated by the Tokyo Electric Power Company (TEPCO) and the simultaneous progression of severe core damage at multiple units and the continuation of large scale releases of radioactivity over an extended period of time. The fact that this accident has raised concerns around the world about the safety of nuclear power generation is a matter that Japan takes with the utmost seriousness.

In the face of this hardship, however, Japan has received a tremendous outpouring of support and expressions of solidarity from around the world. Approaching the second anniversary of the event, I would like to convey the Japanese people's sincere gratitude to the global community for its support.

Immediately after the accident, various accident investigation teams were organized and they published their judgement and lessons learned. Most of them judged that though the accident was triggered by a massive force of nature, it was existing weaknesses regarding defence against natural hazards, regulatory oversight, accident management and emergency response that allowed the accident to unfold as it did. The publication of these judgements angered the public and it became difficult for nuclear power plant operators to obtain the permission of local governments to restart their plants after refuelling and inspection outages. Accordingly, at present only two units are in operation, as the Prime Minister asked the Governer to allow their restart after refuelling and inspection outage.

The Government started the review of the energy policy, including the nuclear energy policy, taking a critical look at the causes and impacts of this accident. Though it should be reasonable to assume at this moment that Japan will decrease its dependence on nuclear power in future, it will take some time to complete the review as the Government changed as a result of a general election held at the end of last year. The possibility of restarting each idling plant in the future will depend on the decision of its operator to make investment to comply to new safety regulations to be introduced by the Nuclear Regulation Authority (NRA). The present paper summarizes the key issues in the deliberation we have started last year on the fast reactor research and development (R&D) programme that should be compatible with the assumption that Japan will reduce dependence on nuclear energy in the long run.

2. FAST REACTOR R&D ACTIVITIES BEFORE 11 MARCH 2011

As Japan is meagerly endowed with economically recoverable natural energy resources, the Japanese Government had put emphasis on the assurance of energy security and the need to minimize dependence on the import of energy resources in the deliberation of its energy policy. Recognizing that nuclear power could meet this need, the Government had been promoting nuclear power generation as one of the main strategic policy options. The main policy elements regarding nuclear power generation had been;

- Continue to construct and operate nuclear power plants as a major element of electricity production;
- Recycle uranium and plutonium recovered from used fuel in light water reactors (LWRs);
- Steadily promote R&D of sodium cooled fast reactors (SFRs) that can improve uranium utilization dramatically.

The first step of Japan's SFR R&D programme was to construct the JOYO experimental fast reactor at Oarai. JOYO had been operating at 50 MW(t) since 1977 and boosted to 100 MW(t) in 1983 and to 140 MW(t) in 2003. It has been shut down since 2007 due to damage of upper core features of the instrumented assembly caused by the malfunction of the fuel handling machine.

The second step was MONJU, the 280 MW€ prototype FBR that was constructed at Tsuruga. Though it started operation in April 1994, a sodium leakage occurred in its secondary heat transfer system during performance tests in 1995. The operator's inappropriate information management in the face of this event gave rise to the loss of public trust in its operator and caused it to be shut down for almost 15 years. It restarted in May 2010. It has three coolant loops, uses MOX fuel, and is designed to produce 714 MW(t) and 280 MW(e).

On the other hand, Tokai Reprocessing Plant (TRP) that recovers plutonium from used LWR fuel was constructed and the MOX fuel for JOYO and MONJU has been fabricated at the Plutonium Fuel Production Facility (PFFF) since 1988.

In 2006, the FaCT Project was started: the objective of this project was to establish a feasible concept of innovative fast reactor and its fuel cycle system to be introduced into the grid by 2050, satisfying the goals for the Generation IV reactor system in terms of sustainability, economics, safety and reliability, and proliferation resistance and physical protection. The team selected the Japan Sodium-cooled Fast Reactor (JSFR) concept for detailed review and identified innovative technologies to be developed for this concept to satisfy the goals. A demonstration fast reactor model based on the JSFR concept was due to be committed in 2015 based on the result of the development of these technologies with a view to putting the reactor on-line in 2025.

3. THE 11 MARCH ACCIDENT AT FUKUSHIMA DAIICHI NUCLEAR POWER PLANT

In March 2011, Units 1–4 of the Fukushima Daiichi plant were seriously damaged in a major accident. The radioactive release from the plant caused radioactive contamination of a wide area of the land around the plant. Hot spots were identified even 260 km from the site in several directions, and about 80 000 people are still displaced as the radiation level of their homes is higher than 20 mSv/year and about the same number of people have chosen to leave home from the fear of exposure to radiation, and though the radiation level of their homes is below 20 mSv/year. Many of them have suffered psychologically due to fear of radiation, separation of family, disruption of communities, etc. Trade of agricultural and marine products harvested in the neighborhood of the plant are still restricted. Damage compensation is estimated to be at least about

6 trillion yen (US $70 billion) at present. It should also be emphasized that though no one has been directly hurt by the radiation exposure itself, the accident has caused several hundred deaths due to the worsening of diseases owing to dislocation from hospitals including emergency evacuation and the stress of life in a shelter after dislocation.

Various accident investigation teams analysed diverse aspects of the cause and the progression of this accident and the way it impacted on people's lives and society as a whole, and recommended actions nuclear operators and the Government should take immediately. Key recommendations related to the prevention and mitigation of a severe accident are as follows:

— A strong safety culture should be established in every nuclear enterprise.
— There should be strong leadership in all the institutions involved in nuclear power that ensures attention to safety, as well as continuing efforts to understand the technology and to improve it.
— Every operator should recognize its fundamental responsibility for safety, continuously striving for safety excellence and making regular investments to address insights arising from operating experience and evolving knowledge of external events.
— A major uncontrolled release of radioactivity, whether driven by natural or malicious causes, is unacceptable as it causes loss of sociopolitical stability needed for nuclear power. Therefore, actions should be taken to ensure that;
  - Design basis external events, including seismic, seismic-tsunami and other events, and their combined effects are properly evaluated.
  - Extended losses of power and ultimate heat sink are covered under severe accident conditions, and protection is provided by a diverse and flexible capability of providing power and cooling.
  - Accident management procedures, including a reliable hardened vent for specific reactor containment, are established to respond to a beyond design basis event, taking into consideration the fact that external events might affect the entire site, and training thereof are in place.
  - Emergency preparedness and response capabilities are in place and available even under combined effects of natural events.
— The regulator must be competent, independent, and dedicated to the task of ensuring that safety obligations are fulfilled.

In response, the Government established the NRA governed by five full-time commissioners in September last year as a new independent regulatory organization that is responsible for nuclear safety, and the security and safeguarding of nuclear materials. The NRA immediately started reviewing the characteristics of active faults in and around several NPP sites that had been open to dispute and also started to establish a new safety standard that requests the implementation of countermeasures against severe accidents that should reflect the recommendations mentioned above by July this year, which will be the criterion for the NRA to allow the restart of idling plants.

4. OBJECTIVES OF NUCLEAR ENERGY R&D AFTER 11 MARCH

The former Government concluded after one year deliberation of future energy policy that Japan should strive to maximize both renewable energy use and efficiency of energy use so as to be able to reduce dependence on nuclear energy, while utilizing existing nuclear power stations as an important source of electricity once the NRA confirms their safety. Now, the new Government announced that it would review energy and environmental strategy again. It will not be necessary for the Japanese nuclear energy community, however, to change its planning base for post-3.11 nuclear energy policy from the assumption that the future capacity of nuclear power generation in Japan will be smaller than that in the past.

Even under this assumption, however, we believe it important for Japan to continue the R&D of fast neutron reactors, exploring innovative fast reactor technologies that will reinforce the global platform for this type of nuclear reactor, as its development and deployment is important to the sustainability, reliability and security of the world's long term energy supply.

Of course, we should assume that the budget for SFR and its fuel cycle technology R&D will shrink for the time being as the priority of nuclear R&D in Japan should be put on the exploration of innovative technologies usable for promoting cleanup and decommissioning of the severely damaged Fukushima Daiichi NPP, as well as for decontaminating areas contaminated by the accident.

We believe that though the expected cost is vast, the first objective of the R&D should be to accomplish the full power operation of MONJU as it will contribute to the establishment of technological basis for a safe, reliable SFR to be located in an area of high seismicity. We are asking the JAEA, the operator of MONJU, to prepare a detailed plan of activities for this accomplishment based on the risk benefit analysis of each activity for attaining this objective, taking into consideration the prerequisite that the operator of nuclear power plants in Japan should take the necessary measures for complying with the NRA's new set of safety standards and establishing procedures for severe accident management and providing training thereof.

The second objective under consideration is to promote the R&D of technology to reduce the volume and toxicity of waste from nuclear power generation utilizing fast neutron reactors. Fortunately, the MOX fuel currently loaded in the core of MONJU contains a significant amount of americium-241, the daughter of plutonium-241 due to delay in fuel loading and restart. Therefore, it is possible to acquire data on an americium loaded core during the initial phase of MONJU operation. We are reviewing the possibility of accumulating knowledge related with the reduction in volume and toxity of waste from nuclear power generation by loading MOX fuels that contain minor actinides, establishing technologies for fabricating such fuels, and executing post-irradiation examination of them and developing technology to reprocess them. In this connection, we will promote activity to repair JOYO in a few years as it is a versatile fast neutron irradiation facility for such developmental activities.

The third objective under consideration is to promote the R&D for strengthening safety features of SFRs, and for prevention and mitigation of severe accident in SFRs that can eliminate the need for preparation of off-site emergency response, in particular. The JAEA has already promoted level 2 PSA utilizing simulation tools such as SAS 4A, SIMMER, and CONTAIN/LMR for core disruptive accidents and extracted information on the features effective to prevent and/or mitigate the severe accident of SFRs. Information has already been focused on the preparation of severe accident management features.

We expect that the JAEA will continue to promote the R&D for a passive shutdown system that utilizes the phenomenon that a magnet loses all of its magnetic properties at Curie temperature, a shutdown cooling system based on natural circulation utilizing atmosphere as an ultimate heat sink, and a containment system that can provide both retention of molten fuel in a reactor vessel and containment of radioactive material within the containment vessel. As these features are not add-on but built-in features of existing safety systems, they will increase the robustness of safety functions realized by such safety systems.

The fourth objective is to support innovation in nuclear energy technology, safety, waste management, non-proliferation and security relevant to fast reactors and its fuel cycle technology. As activities to pursue such innovation provide the basis for the promotion of R&D activities aiming at the aforementioned objectives, we expect the Government and nuclear industry to strengthen coordination and organizational and mission alignment across the country in the promotion of such support from that viewpoint. At the same time, the support should also be recognized as an investment in the human capital and technical expertise needed for sustaining a leadership role in the future, it should involve a broad range of participants, including universities, industry, and national laboratories, in cooperation with international research partners, so as to maintain vital human expertise such as highly trained nuclear scientists, engineers, and workers, etc., as well as critical infrastructure and facilities.

Last but not least is international cooperation. Japan has believed it important to sustain cooperative international efforts with the goal of realizing an innovative fast reactor and its fuel cycle system that will demonstrate globally acceptable characteristics, and has participated positively in various cooperative projects for SFR development, including four cooperative project agreements for SFR within the framework of GIF. Japan has also recognized close international cooperation and contacts between different scientific and technical fields created in such research projects contribute to deepening and broadening know-how among researchers.

Though we believe international cooperation will continue to be an essential and indispensible part of SFR R&D, we will be forced to decide which projects should be participated in so that they maximize the benefits to our objectives, owing to severe budgetary constraints. It has been stressed in the deliberation of post-3.11 energy policy that it is a responsibility of the Japanese nuclear community to contribute to strengthening nuclear safety worldwide by sharing its experience and lessons derived from the severe accident with the world. Keeping this recommendation in mind, we should continue doing our best to facilitate fruitful international cooperative activities in pursuing these objectives.

5. CONCLUSION

The 11 March severe accident at Fukuhsima Daiichi nuclear power plant in Japan has changed the public's perception towards nuclear power and it is now predicted that nuclear power will decrease its contribution to energy supply in the long term. Taking this prediction into consideration, Japan is in the process of deliberating the future R&D strategy for the SFR and its fuel cycle technology. Currently, the following are under discussion as major objectves of R&D for SFR and its fuel cycle technology in the future: (1) to accomplish the full power operation of MONJU, improving its safety feature in accordance with new safety standards to be set by the NRA, (2) to explore innovative fast reactor technology for reducing the amount and toxic level of radioactive waste, (3) to explore innovative technologies for strengthening the safety of SFRs, (4) to support innovation in nuclear energy technology, safety, waste management, non-proliferation and security, relevant to fast reactors and its fuel cycle technology. As these are major objectives of R&D for a sustainable nuclear energy system, Japan will carry out activities to progress these objectives in close cooperation with the international community.

FAST REACTOR DEVELOPMENT PROGRAMME IN THE RUSSIAN FEDERATION

V.I. RACHKOV[*], E.O. ADAMOV[**], A.V. LOPATKIN[**],
V.A. PERSHUKOV[***], V.M. TROYANOV[*]

[*] ITC “PRORYV” Project

[**] N.A. Dollezhal Research and Development Institute
of Power Engineering (OJSC NIKIET)

[***] Rosatom Nuclear Energy State Corporation (Rosatom)

Moscow, Russian Federation

Abstract

Problems of present-day nuclear power development and possible solutions based on the fast reactor technologies and the closed nuclear fuel cycle (CNFC) are analysed in the paper. Technical requirements for the nuclear power systems have been determined for creating large scale nuclear power in the Russian Federation. This report focuses on the objectives and feasible technological goals which should be achieved in the framework of the Federal Target Programme “Nuclear Power Technologies of the New Generation for the years 2010–2015 and Prospects to 2020”, including the “Proryv” Project (“Breakthrough”) elaborated in 2012.

1. THE PROBLEMS OF PRESENT-DAY NUCLEAR POWER DEVELOPMENT

The Russian Federation possesses the full cycle technology of nuclear power engineering: from mining the uranium ore to electric power generation and reprocessing of spent nuclear fuel. At the beginning of 2012, 33 power units with a total installed capacity of 23.6 GW(e) were operated at ten nuclear power plants (NPPs). The basis of present-day nuclear power is formed by thermal reactors — WWER (water cooled, water moderated power reactors) and RBMK (large capacity graphite channel-type reactors) operated in the mode of open nuclear fuel cycle with a partial use of regenerated uranium. The share of nuclear power in the total energy balance in the Russian Federation amounts to 6%, whereas its share of electric power generation is about 17%. Nuclear power is of great importance for the European part of the Russian Federation, in particular, in the northwest, where the NPP generated electricity share reaches 42%.

The milestones of nuclear power development for the middle term up to 2030 are defined in the “Energy Strategy of Russia” approved in 2009 [1]. According to this strategy, the NPP capacity is planned to be increased up to 52–62 GW(e) up to 2030, covering about 30% of the electrical needs of the country. The long term objectives of nuclear power development in the country are published by the Institute for Energy Research of the Russian Academy of Sciences in, “The prospects of NPP development to the middle of the 21st century” issued in 2010. The increase of NPP installed capacities in the Russian Federation, up to 2050, is considered within 100–140 GW(e). It is supposed that by the middle of the century nuclear electricity generation will reach 33–42% in the entire country.

The global economic crisis, which started in 2008, led to a certain slowdown of economic growth in the country and as a result, a reduction of the current energy consumption and its growth rate. The accident at the Japanese NPP Fukushima in March 2011 was one more additional challenge to the planned near-term development of nuclear power.

These two factors had an impact on the near-term plans of nuclear power development in many countries, including the Russian Federation. In the draft plan of the Ministry of Energy published in July 2012, the planned nuclear power growth in the Russian Federation for the period to 2020 has a 15% reduction, from 39–41 GW(e) to 34 GW(e) and now is limited mainly by the plants which are now in the process of construction.

However, a crisis can only cause a temporary reduction of the energy resource consumption and as a consequence, decrease the interest to problems of sustainable long term energy supply. With the inevitable growth of population and industrial production in many countries, it can be expected that the energy and ecological problems will be aggravated, and that under conditions of limited and unevenly distributed resources of hydrocarbons, the interest to nuclear power will be increased again.

It is beyond a doubt that a further development of nuclear power in our country and in the world up to 2030 can be realized only on the basis of the commercially matured technologies of thermal reactors and the open nuclear fuel cycle which meet the safety requirements elaborated earlier. Rosatom is working on the new project “WWER-TOI” (standard optimized and informational project of a double-unit NPP with WWER) with a unit capacity of 1200–1300 MW(e). The present-day safety requirements and trends of the construction industry development were taken into consideration while designing the “WWER-TOI”. The NPPs with such nuclear reactors can provide the nuclear power capacities planned level by 2030, however, it should be admitted that realizing the ambitions of large scale development of nuclear power in the second half of this century on the basis of thermal reactor technologies can become unrealized due to a number of problems inherent with the thermal reactor systems:

- An increasingly serious loss of public confidence after severe accidents at NPPs, whereas the safety improvement of thermal reactors is accompanied by the increase of specific capital costs on new NPPs, overrating those for the thermal power stations;
- With significant amounts of irradiated fuel accumulated in the world, the task of its final disposal is not solved yet and the public considers the problem of "nuclear waste" unsolvable in principle;
- The technologies used at enterprises for uranium enrichment and reprocessing of spent nuclear fuel from thermal reactors with extraction of plutonium poses a risk of proliferation of nuclear weapons materials;
- Economically acceptable reserves of natural uranium, if used ineffectively in the conventional open fuel cycle, are not large, and this can limit the scale and functioning of the present-day nuclear power with the open fuel cycle within the current century.

It is for these problems that, in 2012, one of the most representative systematic investigations of recent years in the nuclear power sphere, the Global Energy Assessment (GEA) project, has provided an extremely unfavourable conclusion for the nuclear power prospects:

> "Nuclear energy is a choice, not a requirement. The GEA pathways illustrate that it is possible to meet all GEA goals even in the case of a nuclear phase-out. Nuclear energy … prospects are particularly uncertain because of unresolved challenges surrounding its further deployment, as illustrated by the Fukushima accident and unresolved weapons proliferation risks".

However, on the other hand, these investigations, like many other earlier national and international system studies, show the possibility and appropriateness of a significant strengthening of the role of nuclear power in sustainable energy supply on the basis of new generation nuclear power technologies with characteristics that meet the challenges of present-day nuclear power.

2. STRATEGY OF THE LARGE SCALE DEVELOPMENT OF NUCLEAR POWER IN THE RUSSIAN FEDERATION BASED ON FAST REACTOR AND CNFC TECHNOLOGIES

In the Russian Federation, as early as the end of the 20th century, experts of several departmental and academic institutes fulfilled a scope of work for defining the main tasks and pathways for the development of large scale nuclear power of this country. The in-depth understanding of the preceding period of nuclear power evolution leads to the comprehension of the priority to be rendered to technological renovation which could be a justification for bringing up the question of the large scale development of nuclear power. Actually, this work produced the development of "The Strategy of Nuclear Power Development in Russia in the first half of the 21st century" (henceforth referred to as "Strategy-2000") approved by the Russian Federation Government in May 2000 [2].

The basis of "Strategy-2000" is formed by the "natural safety" principle, which implies a complex solution to the safety problems in nuclear power founded on the creation of new generation nuclear power technologies, including:

— Excluding the accidents which require evacuation of local population, or, moreover, its relocation, as well as resulting in setting aside considerable areas from economic exploitation;
— Complete use of the energy potentials of raw materials mined;
— Turnover of nuclear materials in the fuel cycle, with the natural radiation balance* preserved;
— Technological reinforcement of the non-proliferation regime for nuclear weapons;
— Ensuring the competitiveness of nuclear power compared with other kinds of energy generation.

These provisions were at the centre of the Russian Federation President's Initiative at the Millennium Summit in the UNO (New York, 6 September 2000) for the use of nuclear power in sustainable development in the new millennium. It should be kept in mind that in the aspect of nuclear power perception by the society, the 1990s have some features in common with the contemporary

* The preservation of natural radiation balance implies that after a certain period, not long in historic scales, total radioactivity and radiotoxicity of the wastes produced as a result of NPP operation and reprocessing of irradiated fuel and dispatched for final disposal will not exceed the total radioactivity and radiotoxicity of isotopes of the uranium raw materials extracted from the earth's crust for the NPP fuel supply.

post-Fukushima period in the situation of a continuous global economic crisis. The Russian “Strategy-2000” was the first State level document, which contained some emerging evidence of changed attitudes to the use of nuclear power in the world. Later on, at the beginning of the next century, two international projects were launched — the Generation-IV International Forum and INPRO (IAEA), with similar objectives — which characterize the challenges to global nuclear power development in various areas (safety, non-proliferation, wastes, resource basis, and economy), and develop the requirements and methodology for evaluating the prospects of different nuclear power systems.

The “Strategy-2000” also determines several technical requirements stated for large scale nuclear power. In particular, as applicable to reactors, these goal-oriented requirements include the following:

— Ensuring of a minimal reactivity margin in the reactor, which would not allow a realization of the prompt-neutron runaway;
— Preservation of the design basis physical characteristics during the campaign in a combination with self-regulation effects;
— Cancelling the use of primary circuit coolant under high pressure creating the risk of loss of the circuit tightness and cessation of reactor cooling, which can cause inadmissible releases of radionuclides;
— Changeover to the liquid metal coolant with a high boiling point and the integral shaft design of the reactor, eliminating the possibility of loss of cooling;
— Cancelling the use of coolants and materials typified by chemical activity and flammable in the interaction with air and water.

As applicable to plants and technologies of CNFC, the target-oriented requirements elaborated in the “Strategy-2000” include the following:

— Cancelling the direct burial of irradiated fuel, minimizing the quantities of high-level wastes in the CNFC, with hazardous long lived actinides and a certain part of fission products returned to the reactors in the composition of regenerated fuel;
— Closure of nuclear fuel cycle with minimized cooling time for the irradiated fuel and reduced quantities of nuclear materials subject to storage;
— Decreasing the risk of nuclear materials proliferation, with refusal from breeding plutonium in the fast reactor blankets and separation of pure plutonium from irradiated fuel, as well as resulting from a changeover to operations with “dirty” fuel and elimination of long range transportation of irradiated fuel and nuclear materials.

Obviously, large scale nuclear power can be developed only in the case that it is competitive in the energy production and investment market, with gradual retirement from a universal practice of State support. The principle of natural safety serves for this objective as well. The consecutive implementation of this principle, beginning with engineering solutions aimed at the simplification of structures, reduced strictness requirements to the equipment, buildings, site, and to the personnel would allow realizing the economic approach, "the safer, the cheaper".

It was unambiguously suggested by the "Strategy-2000", that the requirements formulated for large scale nuclear power can be met only on the pathway of the fast reactor and CNFC technologies.

The conclusion about an important role of fast reactor technology and CNFC for large scale nuclear power is not a new or unique one. In the 1960s to 1980s, many industrialized countries (France, Germany, Italy, Japan, Sweden, the former USSR, the United Kingdom and the United States of America) had their own national programmes for the development of fast reactors and CNFC that, unfortunately, did not provide any commercial results. One of the main reasons for this was the high cost of NPPs employing sodium cooled fast reactors (compared with PWRs). Another reason was in the low prices of natural raw materials, which is now only a reminiscence of the past.

The international assessments of different nuclear power technology systems in the framework of the Generation IV International Forum and the IAEA INPRO projects carried out at the beginning of this century also pointed out the principal advantages of CNFC with fast reactors for resolving the tasks of large scale nuclear power development.

Nowadays, practically all countries with large scale nuclear power programmes (France, Japan, the Republic of Korea and the USA to a certain extent), either have their national programmes for mastering the fast reactor based CNFC systems, or plan its full-scale use in the future (China, India).

A national programme for the deployment of fast reactors and CNFC technology had existed in the former USSR for many years. Two sodium cooled fast reactors were designed in the 1970s and 1980s: BN-350 and BN-600, followed by the BN-800 project and construction of a MOX fuel fabrication facility for fast reactors at PA "Mayak".

After the Chernobyl accident in the period of national economy reformation, the scope of works in the area of high technology development, including the fast reactor and CNFC, was substantially reduced. However, by the time of revival of the nuclear industry in the 2000s, a high level of designing expertise for the fast reactor technologies and CNFC was still available in the country.

The operation of the BN-600 reactor at Beloyarsk NPS is worth a special note. This reactor has been in operation for 32 years, without any serious problems. Nowadays, it is the only operating fast reactor in the world. The justification for an operation time extension from the designed 30 year period to 45 years has been worked out for this reactor.

In addition to successful operation of sodium cooled fast reactors, long term experience of lead–bismuth cooled reactor facilities for nuclear submarines has been gained in the country. On the basis of this experience, the SVBR-100 fast reactor project with 100 MW(e) capacity for multipurpose use in regional nuclear power is developed nowadays in the framework of a State–private partnership.

The BREST project of lead cooled fast reactor with on-site nuclear fuel cycle is also developed in the country.

The following technologies in the area of CNFC have been developed and demonstrated at the pilot production or experimental levels:

— Industrial water technology for reprocessing spent nuclear fuel from vehicle reactors, WWER-440 and BN-600 at RT-1 plant;
— Experimental technology of MOX fuel production, provided the possibility for testing about 50 full scale fuel assemblies with pellet fuel, and about 30 vibro-compacted fuel assemblies in the BN-350 and BN-600 reactors;
— Experimental technology of nitride uranium fuel production for the BR-10 reactor operated on this type of fuel for 18 years;
— Laboratory technology of pyrochemical irradiated nuclear fuel processing.

The results of assessment of various nuclear power systems in terms of “natural safety” criteria have revealed the highest potential of compliance with these criteria for the system that includes a fast reactor with nitride fuel and lead coolant with the on-site nuclear fuel cycle using dry methods of spent nuclear fuel reprocessing.

However, the main problem of this system is quite obvious: as of today, none of the technologies involved in this system have been demonstrated at the pilot–industrial level in respect of performance capabilities.

3. FEDERAL TARGET PROGRAMME “NUCLEAR POWER TECHNOLOGIES OF A NEW GENERATION”

A new phase of the CNFC technology development with fast reactors began in the Russian Federation with the preparation of an R&D programme approved by the Russian Federation government in 2010 with a new technological nuclear power platform under the Federal Target Programme “Nuclear Power

Technologies of the New Generation for the years 2010–2015 and Prospects to 2020", as well as the Project of the Commission of the Russian Federation President aimed at the modernization and technological development of the Russian economy, "New Technological Platform: Closed Nuclear Fuel Cycle and Fast Neutron Reactors".

The goal of the programme is to develop nuclear power technologies of a new generation based on fast reactors with CNFC for nuclear power plants to meet the demands of the country for energy resources and ensure an improved efficiency in the use of natural uranium and spent nuclear fuel.

Total amount of funds allocated to the implementation of the programme in accordance with this scenario covers about 100 billion rubles. Due to a long duration of its implementation period and the highest priority to get the principally new technical solutions on fast reactor technologies, the programme is implemented in two phases.

During the first phase (2010–2014) the following results will be achieved:

— Gaining technical solutions of principle novelty and the development of new advanced engineering designs of power units with fast neutron reactors with lead, lead–bismuth, and sodium coolants;
— Completion of the design works and commissioning of the fuel production complexes to provide the MOX fuel for fast neutron reactors;
— Preparation of the detailed design documentation for the project for construction of the multi-purpose fast research reactor (MBIR).

During the second phase (2015–2020) the following results will be achieved:

— Construction of a demonstration prototype power unit with a fast reactor cooled by lead, and a commercial prototype power unit with a fast reactor cooled by lead–bismuth, and a multi-purpose fast research reactor (MBIR) to replace the BOR-10 reactor currently in operation;
— Commission and put into operation a set of technically refurbished large physical test facilities;
— Construct and put into operation an industrial complex for dense fuel fabrication for fast reactors;
— Create a multi-functional radiochemical R&D complex with the aim of developing and mastering advanced CNFC technologies;
— Complete the development and justification of technological design solutions for the on-site industrial module for reprocessing spent nuclear fuel from fast neutron reactors;

— Accomplish the development and justification of technological design solutions for a specialized line of the centralized plant for reprocessing of spent nuclear fuel from fast and thermal reactors.

In 2012, the “Proryv” Project was launched within the framework of the Federal Programme “Nuclear Power Technologies of the New Generation for the years 2010–2015 and Prospects to 2020”, consolidating the projects dedicated to the development of large capacity fast reactors, CNFC technologies, and new types of fuel and material. It is aimed at the achievement of nuclear power of a new quality level. The goal of the project is to develop and construct a nuclear power complex that includes NPPs with fast reactors, a production site to regenerate (reprocess) and refabricate nuclear fuel, and a site to prepare all types of waste for the final disposal from the process cycle, with the whole complex meeting all the natural safety requirements.

As a result, the following will be implemented:

— Pilot industrial production of nitride fuel (MNIT);
— R&D, design and construction of a demonstration prototype power unit with the BREST-OD-300 reactor, in compliance with the natural safety requirements;
— R&D, design and construction of on-site nuclear fuel cycle facility for the demonstration prototype power unit with BREST-OD-300;
— R&D and design of the BN-1200 reactor in compliance with the natural safety requirements;
— R&D and design of separate CNFC elements in the form of a specialized line of the centralized plant;
— The design of the first-of a kind power complex that meets the natural safety requirements and includes NPPs with fast neutron reactors of 1200 MW(e) capacity and on-site nuclear fuel cycle, to be constructed in 2025.

4. CONCLUSIONS

— Large scale nuclear power can be only created with implementation of all the “natural safety” principles;
— Adequate technological elements are necessary to be formed for large scale nuclear power development in the second half of the XXI century;
— Large scale nuclear power can be only developed on the basis of the CNFC technologies with fast reactors;

— In the Russian Federation, the alternative technologies (BREST and BN, MNIT and MOX) and nuclear fuel cycle arrangement systems (on-site nuclear fuel cycle and centralized nuclear fuel cycle) are assumed to be developed under the “Proryv” Project launched in 2012, so that by the year 2020 the best technological and economically competitive options should be chosen for the CNFC system with fast reactors for large scale nuclear power.

REFERENCES

[1] MINISTRY OF ENERGY OF THE RUSSIAN FEDERATION INSTITUTE OF ENERGY STRATEGY, Energy Strategy of Russia for the Period up to 2030, Approved by the Order of RF Government No 1715-p of November 13, 2009, Moscow (2010).

[2] MINISTRY OF ATOMIC POWER ENGINEERING OF THE RUSSIAN FEDERATION, Strategy of Nuclear Power Development of Russia for the First Half of the XXI Century, Basic Provisions. Approved by the RF Government of 25.05.2000, Minatom, Moscow (2000).

US RESEARCH PROGRAMME TO SUPPORT ADVANCED REACTORS AND FUEL CYCLE OPTIONS

P.B. LYONS
US Department of Energy, Office of Nuclear Energy,
Washington, DC,
United States of America

Abstract

Nuclear energy is an essential element of the United States of America's clean energy portfolio. The US Department of Energy's Office of Nuclear Energy is addressing the challenges to the expansion of civilian nuclear energy through a science based research and development (R&D) programme. This paper describes our research in the areas of advanced reactor technology and fuel cycle options, and our engagements with key stakeholders including international partners, private industry, the regulator and the university community.

1. INTRODUCTION

All governments share a common challenge to ensure their people have access to affordable, abundant, and environmentally friendly sources of energy. Events at Fukushima have reinforced all of our efforts towards developing even safer and more robust nuclear technologies. Light water reactor (LWR), fast reactors, and other advanced reactor concepts provide opportunities to incorporate inherent and passive safety features in new nuclear reactor designs.

Over the past few years, the United States of America has put considerable effort into addressing the issue of nuclear waste and the back end of the fuel cycle. In January 2010, the Blue Ribbon Commission on America's Nuclear Future (BRC) was tasked by President Obama to conduct a comprehensive review of America's nuclear waste problem.

The BRC provided eight recommendations in its final report [1] in January 2012 and the Administration released its Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste [2] outlining a path forward based on the Commission's recommendations in January 2013. The Department of Eneregy (DOE) will focus this year on the development of a path forward to address this issue in partnership with the US Congress and other key stakeholders. Future US fuel cycle options, nuclear waste disposal and potential commercialization of fast reactors or reprocessing technologies are intimately linked.

President Obama is fully supportive of expanding clean energy technologies as part of an “all-of-the-above” approach to meet our energy and environmental needs. For example, in the Republic of Korea last March, he stated, “With rising oil prices and a warming climate, nuclear energy will only become more important. That’s why, in the United States, we’ve restarted our nuclear industry as part of a comprehensive strategy to develop every energy source.” Nuclear power has reliably and economically contributed almost 20 percent of electrical generation in the USA over the past two decades. It remains the single largest contributor (more than 70%) of non-greenhouse-gas-emitting electric power generation in the USA. Currently, there are five commercial nuclear power plants under construction in the USA, including four AP1000 reactors, which represent a new generation of passively safe nuclear plants.

The Obama Administration has recognized the importance of nuclear energy and supports R&D programmes that can help to expand the benefits of nuclear energy. One example of this commitment is our new programme to support small modular reactors (SMRs). As part of a five-year US $452 million programme, we selected one industry partner to cost-share a licensing demonstration of SMR technology. A funding opportunity announcement for additonal awards may be made in the future.

2. NUCLEAR ENERGY R&D PROGRAMME

Our primary mission is to advance nuclear power as a resource capable of making major contributions to meeting the nation’s energy supply, and environmental and energy security needs. In April 2010, we issued a Report to Congress [3] outlining a “roadmap” for our nuclear energy research programme. The roadmap focuses on four objectives that address the challenges to expanding the use of nuclear power:

(1) Understand ageing phenomena in operating reactors and develop technologies that improve the reliability, sustain the safety, and extend the life of current reactors;
(2) Conduct R&D on new reactor technologies, including improvements in the affordability of new reactors to enable nuclear energy to help meet the Administration’s energy security and climate change goals;
(3) Develop sustainable nuclear fuel cycles;
(4) Understand and minimize the risks of nuclear proliferation and terrorism.

Consistent with the theme of this conference, this paper focuses on two of those goals: improving the affordability and mission space of new reactors to expand the benefits of nuclear energy, and; developing sustainable nuclear fuel cycles.

2.1. Advanced reactor technologies R&D

We are working to develop advanced reactor and fuel cycle technologies that further enhance safety, economics, performance, sustainability, environmental impacts and proliferation resistance. In addition to our domestic activities, we are leveraging our efforts through several bilateral and multilateral international partnerships to extend the benefits of clean, safe, civilian nuclear power globally. International cooperation is essential to share expertise, leverage scarce resources, and help to assure that nuclear energy remains safe, reliable, economical and, most importantly, acceptable to the public.

Fast reactor technology was conceived early in the nuclear era as a means to fully utilize the energy content of uranium resources by increasing efficiency through converting and consuming actinides, which ultimately reduces high level radioactive waste. The USA was the pioneer of fast reactor technology. This effort included work on several major sodium cooled fast reactor (SFR) facilities and demonstration programmes. Examples include:

- Design and construction of the Experimental Breeder Reactor I (1951–1964) test reactor with the first demonstration of electricity from nuclear power in 1951.
- Design and construction of the unique pool configuration of the Experimental Breeder Reactor II (1964–1994) with demonstration of fuel cycle closure in 1964–1969.
- Design and construction of the Fermi-1 (1963–1972) commercial power reactor.
- Design and construction of the Fast Flux Test Facility (1980–1991).

Although there are no current US fast reactor or reprocessing demonstration projects, the DOE is pursuing R&D in several areas to advance the state of the art, maintain critical technical capabilities, and address cross-cutting technologies that will support future fuel cycle options.

The US fast reactor programme is focused on long term, science based R&D that supports increasing the performance of fast reactor technology. This increased performance can include increasing the safety, reducing the cost, increasing the electrical power output (efficiency) and developing technologies for improved operation or maintenance of fast reactor systems. Additionally, we are working on advanced materials, inspection technologies, advanced energy conversion systems, advanced compact reactor concepts, advanced fuel handling systems and advanced modelling and simulation code development.

We recognize that reducing capital costs is a key factor in future reactor deployment, consistent with the second objective of the R&D roadmap. We are pursuing multiple pathways to reduce capital costs, including design simplification, commodity reduction, advanced energy conversion, and improved material performance. Concept development studies have been initiated and a wide variety of system innovations have been considered. These efforts include examination of advanced systems and components such as compact fuel handling mechanisms, advanced balance of plant systems, ultra-long lived fast reactor cores and advanced heat exchanger technology options. To support these activities, we are constructing a metal coolant test facility referred to as the Mechanism Engineering and Testing Laboratory at the Argonne National Laboratory. It will test fast reactor components in a sodium environment. At one time, the USA had a substantial sodium component testing infrastructure available to government researchers and industry. However, that infrastructure has deteriorated substantially and investment is needed to maintain access to critical infrastructure and research capabilities, as well as to support knowledge retention. It is also important for us to cooperate with the international community to leverage infrastructure in mutually beneficial ways.

To further advance fast reactor technology, we are researching advanced materials at locations such as Oak Ridge National Laboratory in order to provide benefits in areas of safety, costs, power output, temperature capability and efficiency. The material R&D programme is closely examining two metal alloys (NF709 and modified grade 92) for sodium compatibility and manufacturability testing, which will be further developed into an American Society of Mechancial Engineers code case.

One of the challenges of typical fast reactor coolants is that they are opaque, further complicating many inspection, maintenance or repair activities. To address this issue, the DOE is funding Pacific Northwest National Laboratory and Argonne National Laboratory for development of under-sodium viewing technology which could be essential for in-service inspection of SFRs. The DOE is pursuing both ultrasonic wave guide and transducer based methods for this purpose.

Another important focus area is modelling and simulation. We are applying high performance computing and simulation techniques towards the creation of virtual SFRs that will enable future designers to optimize their designs for safety and economics. DOE is working on fuel, thermohydraulics and coupled physics models to support this virtual reactor technology.

We see great promise in the supercritical CO_2 Brayton conversion cycle that offers a new, high efficiency energy conversion system capable of long term, high temperature operation. This technology development at Sandia National Laboratories on the secondary systems of new plants can potentially improve efficiency by up to 50% and reduce size by 90%.

2.2. Advanced fuel cycle R&D

In addition to advanced reactor technologies, we engage in advanced fuel cycle R&D to support our roadmap objective: development of a sustainable fuel cycle. Our Advanced Fuel Cycle R&D Program focuses on three broad areas that address both near term and longer term considerations (to around 2050).

These areas are:

- Next generation LWR fuel with enhanced performance and safety and reduced waste generation;
- Metallic transmutation fuels with enhanced proliferation resistance and resource utilization;
- A cross-cutting area of national laboratory equipment and testing capability development to support our science based approach to fuel development.

Our current strategy embodied in our roadmap envisions activities that could demonstrate integration between the LWR and fast reactor systems by mid-century if funded and pursued. The strategy also includes intermediate milestones that require development both in the fuel cycle and the fast reactor development areas.

Additional work performed under the Advanced Fuel Cycle R&D Program, which is also critical for the development of a sustainable fuel cycle, examines technologies relevant to open and closed fuel cycles.

Our efforts to improve the sustainability of a once-through approach to used fuel management begins with increasing the burnup of the fuel — the amount of energy that can be extracted from fuel in the reactor — which may also have the effect of consuming more actinides in the fuel, leaving less to be disposed of. Increasing the burnup of a fuel will require ensuring that both the fuel itself and the structural material designed to keep it in place in the reactor will be able to withstand extended irradiation in the reactor and maintain its integrity when being stored after removal. Deploying advanced fuels will require that they first undergo a qualification process as researchers must irradiate and conduct examinations on test samples to assure their performance. The next generation of LWR fuels will have enhanced performance and safety and reduced waste generation.

We are also developing technologies to allow repeated recycling of transuranic elements in fast spectrum reactors. In the near term, we are working on a fuel cycle options study which will serve to down-select full recycle technologies for further development. In the longer term, after the most promising options are identified, we hope to begin fuel testing for full recycle.

Finally, we are exploring the potential for resuming transient testing within the USA. A robust transient testing capability is crucial for testing the new nuclear fuels presently under development. Existing, non-domestic transient testing capabilities lack in situ, real time imaging technology to determine the behaviour of experimental fuels during transient tests, which is considered crucial in the development, and ultimate licensing, of safer and more efficient fuels and associated materials. The DOE is nearing completion of its analysis of alternatives for resuming transient testing, and plans to move forward with an environmental assessment of alternatives in the very near future.

2.3. Used fuel disposition

We are also undertaking a used fuel disposition campaign which seeks to identify alternatives and conducts scientific research and technology development to enable storage, transportation, and geological disposal of used nuclear fuel and wastes generated by existing and future nuclear fuel cycles. We are performing myriad activities to evaluate the steps needed to move used nuclear fuel from reactor sites to an interim storage facility or a geological repository in the future. In the near term, we are conducting testing of high burnup used nuclear fuel (UNF) cladding properties and canister performance as well as developing a technical basis for licensing transportation systems designed to transport high burnup UNF. The transportation R&D focuses on ensuring transportability of UNF following extended storage, addressing data gaps regarding nuclear fuel integrity, retreivability and demonstration of subcriticality. The disposal R&D focuses on identifying multiple viable geologic disposal concepts in various host media (e.g. mined repositories in salt, clay, shale, granitic rocks and deep borehole disposal in crystalline rock). The recommendations of the BRC and the Administration’s strategy point to an initial focus on interim storage and a geological repository before any possible future shift to more advanced fuel cycles.

3. INTERNATIONAL ENGAGEMENT

Achieving the R&D objectives in the roadmap simply cannot occur without effective international collaboration. As noted earlier, the USA no longer has the full suite of facilities to conduct the needed research. In these times of constrained funding, we need to leverage our financial resources, as do other countries. Nor does the USA have a monopoly on the technical expertise in all of the areas needed to address the challenges facing the long term sustainability of nuclear power. While current US attention is on R&D, we have placed great importance on international collaboration both on a bilateral and a multilateral basis.

Bilaterally, the DOE has comprehensive engagements with such countries as the Russian Federation and France for many technical activities relevant to this conference. With the Russian Federation, the DOE has a broad cooperative programme with Rosatom that encompasses many advanced reactor and fuel cycle topics. A high priority for our cooperation is to support development of the Multi-Purpose Fast Research Reactor (MBIR) as an international user facility. We are also engaged with the French Atomic Energy and Alternative Energies Commission (CEA) to support its efforts on the ASTRID reactor, including joint analysis and benchmarking of transient analysis and neutronics.

In addition, we established a very comprehensive framework of cooperation with Japan under a bilateral commission. The USA–Japan Civil Nuclear Energy Working group includes activities in reactor technologies, fuel cycle technologies and waste management.

Likewise, the USA has a bilateral civil nuclear energy cooperative action plan with China in the area of advanced nuclear technologies. A fast reactor technology working group has been very active in the areas of reactor core design and analysis, reactor plant safety analysis, and the sharing of information on fast reactor technology and operations.

With the Republic of Korea, our Argonne National Laboratory is supporting the Korea Atomic Energy Research Institute (KAERI) in the development of a fast reactor design. In April 2011, the USA and the Republic of Korea launched the Joint Fuel Cycle Studies, which include work related to electrochemical recycling, safeguards and security and fuel cycle alternatives. Phase I of the work, which comprised evaluation of laboratory scale feasibility of electrochemical recycling, was recently completed.

On the multilateral stage, the DOE has an ongoing arrangement for extensive trilateral cooperation with France and Japan on SFRs and is completing the benchmark of EBR-II inherent safety tests as part of an IAEA coordinated research project. The DOE participates in the OECD Nuclear Energy Agency's benchmark activities, such as reactor physics and criticality safety as well as other international activities, to promote safe advanced nuclear technologies.

Similarly, the Generation IV International Forum (GIF) was established in 2000 to carry out cooperative R&D needed to support the next generation nuclear energy systems. The USA, along with China, Euratom, France, Japan, the Republic of Korea and the Russian Federation are actively involved in the GIF SFR projects. As GIF has now entered its second decade, a strategic planning exercise is under way. The original December 2002 technology roadmap is being updated to reflect current advances in fuels, materials, modelling, fuel cycle strategies and lessons learned from Fukushima. Teams are also developing recommendations to further strengthen R&D collaboration and improve ties with other international organizations.

The DOE continues to study the accident at Fukushima using modern severe accident analysis codes with the end goal of comparing different codes from around the world and enabling code improvements and possible safety advances in operating reactors. Another element of the Department's response to Fukushima involves a programme dedicated to development of fuel systems offering improved tolerance to accident conditions.

Consistent with the direction we have received from the US Congress, we have given priority to activities aimed at the development and near term qualification of meltdown resistant, accident tolerant fuels that would enhance the safety of present and future generations of LWRs. We are working to improve the scientific knowledge basis for understanding and predicting fundamental nuclear fuel and fuel cladding performance in nuclear power plants. This knowledge base will be applied to the development of high performance, high burnup fuels with improved safety, cladding, integrity and economics.

Although this work is on the existing LWR open fuel cycle (with no post-irradiation processing using direct disposal) with no direct near term integration with the fast reactor programme, it will be applicable in the longer term if a fast reactor system is integrated into our national energy picture to utilize a closed fuel cycle. If a closed fuel cycle is used in the USA in the future, the advanced LWR fuels will have to be compatible with the closed fuel cycle and its associated waste disposal responsibility. In that case, the initial fuel for fast reactors would come from separated used LWR fuel with successive reloads made from used fast reactor fuel.

4. ENGAGEMENT WITH INDUSTRY AND THE REGULATOR

We are also reaching out to industry to advance fast reactor technologies. Early in 2012, we sought detailed information on advanced reactor concepts from reactor vendors in order to identify their R&D needs. Through the technical review panel process, a team of reactor experts provided advice on eight viable advanced concepts, which included five fast reactor designs. The results of this study will be used to align our R&D investments and we anticipate funding cost-shared R&D projects led by industry to address some of the technology development needs. Priority research needs were identified for sodium-, lead- and gas-cooled fast reactor concepts.

Of course, safety is paramount. In the USA, the Nuclear Regulatory Commission (NRC) is responsible for evaluating the safety of commercial nuclear technologies through the licensing process. The current US regulatory framework for licensing nuclear technologies is based primarily on LWR technologies. As such, the technical review panel identified the need for a licensing framework for advanced reactors in the USA.

Despite a number of advanced reactor concepts being developed by industry, in the USA there is little utility interest in actively pursuing commercial deployment of advanced reactors. Absence of commercial interest in licensing and deploying advanced reactor technologies means there will continue to be considerable risk and uncertainty in the regulatory process. Therefore, we are working with the NRC to facilitate improvements to the regulatory process to accommodate advanced nuclear technologies and with industry to help develop technologies that meet future market needs.

5. NUCLEAR ENERGY UNIVERSITY PROGRAMME AND KNOWLEDGE MANAGEMENT

While our national laboratories and industrial entities are making strides towards potential long term deployment of fast reactors, we continue to invest in education through our Nuclear Energy University Program (NEUP). We provide up to 20% of our R&D funding each year to US educational institutions. Significant elements of this research directly support projects of benefit to future fast reactors. Since 2009, the DOE has awarded US $233 million to universities though the NEUP. We encourage foreign universities to work cooperatively with US institutions to further leverage resources in the US and abroad. We are also pursuing knowledge management programmes to ensure past knowledge and experience is documented in means that will be retrievable as the DOE moves to have greater engagement with the next generation of nuclear experts.

6. CONCLUSION

While the USA is focused on deployment of a geological repository for at least a few decades, we recognize the potential that fast reactor systems may possess. We also recognize that several countries have made their decision to utilize a closed fuel cycle in the near term. Both to assist the international community in assuring safe fast reactor development and in recognition of our possible future needs, we are maintaining R&D programmes that examine fast reactor technology and fuel cycle options, and we pursue this work with international partners, private industry, the regulator and the university community.

REFERENCES

[1] HAMILSTON, L.H., SCOWCROFT, **B.**, AYERS, M.H., BAILEY, V.A., CARNESALE, A., DOMENICI, P.V., EISENHOWER, S., HAGEL, C., LASH, J., MACFARLANE, A.M., MESERVE, R.A., MONIZ, E.J., PETERSON, P.F., ROWE, J.W., SHARP, P., Blue Ribbon Commission on America's Nuclear Future: Report to the Secretary of Energy, Blue Ribbon Commission on America's Nuclear Future (BRC), Washington, DC (2012).

[2] CHU, S., Strategy for the Management and Disposal of Used Nuclear Fuel and High-Level Radioactive Waste, U.S. Department of Energy, Washington, DC (2013).

[3] US DEPARTMENT OF ENERGY OFFICE OF NUCLEAR ENERGY, Nuclear Energy Research and Development Roadmap Report to Congress, Washington, DC (2010).

EURATOM CONTRIBUTIONS IN FAST REACTOR RESEARCH PROGRAMMES

T. FANGHÄNEL, J. SOMERS
European Commission, Joint Research Centre,
Institute for Transuranium Elements
Kalrsruhe, Germany/Ispra, Italy

Abstract

Fast reactors are the basis for a long term sustainable nuclear energy policy, as has been recognized at European level within the Strategic Energy Technology (SET) Plan. On the initiative of the Sustainable Nuclear Energy Technology Platform (SNE-TP), a SET Plan industrial initiative named ESNII (European Sustainable Nuclear Energy Industrial Initiative) gathers the main research and industry players in the field. The Euratom Treaty provides a framework for all civil nuclear activities and the Euratom Framework Programmes provides financial support to several European research projects, amongst them fast reactor programme initiatives. Investigations cover the fields of advanced fuel concepts, their safety relevant properties and irradiation behaviour, as well as separation strategies for closing the fuel cycle and training and education programmes. The results are integrated, at the international level, within the Generation IV International Forum activities.

1. INTRODUCTION

Energy is one of the grand global challenges of the 21st century. People's well-being, industrial competitiveness and the overall functioning of society are dependent on safe, secure, sustainable and affordable energy. On 15 December 2011, the European Commission adopted the communication Energy Roadmap 2050 [1]. The EU is committed to reducing greenhouse gas emissions to 80–95% below 1990 levels by 2050 in the context of necessary reductions by developed countries as a group. Estimates indicate that we are currently around 15–17% below the 1990 level [2].

The challenges posed by delivering the EU's decarbonization objective while at the same time ensuring security of energy supply and competitiveness require developing a long term European framework together with all stakeholders. High public and private investments in R&D and technological innovation are crucial to speeding-up the commercialization of all low carbon solutions.

The power sector has the biggest potential for cutting CO_2 emissions by 2050. To achieve this, the power generation system would have to undergo structural change. All scenarios show that electricity will have to play a much greater role than now and will have to contribute to the decarbonization of transport and heating/cooling.

The EU currently produces almost one third of its electricity from nuclear fission. It represents an important factor in maintaining European competitiveness and the security of energy supply and is an essential component in addressing the challenging needs for greenhouse gas emission reduction. After the Fukushima accident in 2011, the main emphasis in terms of public perception of nuclear energy is now clearly linked to safety aspects.

The scenario analysis of the EU Energy Roadmap 2050 shows that nuclear energy contributes to lower system costs and electricity prices. As a large scale, low carbon option, nuclear energy will remain in the EU power generation mix. Safety costs and the costs of decommissioning existing plants and disposing of waste are likely to increase. New nuclear technologies could help to address waste and safety concerns. The highest safety and security standards need to be further ensured in the EU and globally, which can only happen if competence and technology leadership is maintained within the EU. The world market for the second part of this century will be characterized by the deployment of new nuclear energy systems, on the one hand for increasing the long term sustainability of this energy source and reducing waste quantities, on the other hand for developing new sectors such as hydrogen production, synthetic products or desalination. On the European level, the EU should contribute directly to scientific projects and research and demonstration programmes, building on the Strategic Energy Technology (SET) Plan [3] and the next Multiannual Financial Framework, and in particular Horizon 2020 [4], to invest in partnerships with industry and Member States to demonstrate and deploy new, safe, secure and highly efficient energy technologies.

Owing to the long time periods involved with the storage of irradiated materials, and the associated uncertainties, implementing solutions for the treatment and final disposal of nuclear wastes remains an equally important factor for obtaining the public acceptance of the use of this source of energy. In Europe, two main spent fuel strategies are being implemented, namely the fuel reprocessing followed by the recycling of residual uranium and plutonium in light water reactors, and the direct disposal of the irradiated fuel elements. Both strategies need the deep geological disposal technology for disposing of, either the glass packages issued from the reprocessing, or the fuel elements. Concrete projects are being implemented in several European countries, with the most advanced being in Finland and Sweden.

2. THE EURATOM FRAMEWORK PROGRAMMES

Each European Member State is responsible for its nuclear energy policy, including the back end of the cycle. Thus, R&D programmes are performed on a national level. In support of these, research actions are undertaken under the Euratom research and training framework programmes. Although Member States retain most competencies in energy policy, whether based on nuclear or other sources, the Euratom Treaty has achieved an important degree of harmonization at the European level. It legislates for a number of specific tasks for the management of nuclear resources and research activities. The Euratom Treaty provides a framework for all civil nuclear activities, from the nuclear fuel cycle to industrial and medical applications, the control of foods, and safeguards inspections of nuclear materials. It also addresses environmental concerns such as reducing radioactive emissions, responsible waste management and the promotion of sustainable systems. The Euratom acquis is implemented in strict compliance with the principle that it is the choice of each and every Member State to decide whether or not to use nuclear energy in its energy mix. In this context, the Euratom programme enables the EU to develop, in the interest of all its Member States, the most advanced legal framework for nuclear energy, meeting the highest standards of safety, security and non-proliferation as re-strengthened by European and world leaders at the Nuclear Security Summit 2012 [5], held in Seoul.

In FP7 Euratom [6] there are two associated specific programmes, one covering indirect actions [7] in the fields of fusion energy research and nuclear fission and radiation protection, the other covering direct actions [8] in the nuclear field undertaken by the Commission’s Joint Research Centre (JRC).

FP7 Euratom aims to address the major issues and challenges in nuclear research and to contribute to the further consolidation of the European research area in the nuclear energy sector. It also supports existing community policies while at the same time responding to new policy requirements. This framework programme enhances the creation of a critical mass and new structures in key research fields at the European level as well as the promotion of the free movement of ideas, knowledge and researchers. In general terms, the Euratom research programme aims to develop and assemble knowledge and to improve scientific and technical competences and know-how in support of the safety, security, reliability, sustainability and cost effectiveness of nuclear energy.

With the objective of enhancing in particular the safety performance, resource efficiency and cost effectiveness of nuclear fission and other uses of radiation in industry and medicine. Activities of the indirect actions managed by the Research and Innovation DG include: management of radioactive waste; reactor systems; radiation protection; infrastructures; human resources, mobility and training.

The second programme covering the activities of the JRC in the field of nuclear energy, include: nuclear waste management and environmental impact; nuclear safety; nuclear safeguards, non-proliferation and security. The JRC, initially established by the Euratom Treaty, has since become a leading institution for nuclear research in Europe

3. ADVANCED REACTOR SYSTEMS PURSUED IN EUROPE

Fast reactors are the basis for a long term sustainable nuclear energy policy, as has been recognized at European level within the SET Plan. On the initiative of SNE-TP (Sustainable Nuclear Energy Technology Platform www.snetp.eu), a SET Plan industrial initiative named ESNII (European Sustainable Nuclear Energy Industrial Initiative) [9] has been launched. It gathers the main research and industry players in the field. Three types are followed (which are all three in coherence with the roadmap of the Generation IV International Forum (GIF), namely those cooled with sodium (SFR), lead or lead–bismuth (LFR) and gas (GFR).

Several demonstration projects are now proposed in Europe. ASTRID (SFR) in France and MYRRHA (Pb–Bi cooled accelerator driven system (ADS)) in Belgium are at present the most promising ones, being financially supported by their respective host countries, but LFR (ALFRED) and GFR (ALLEGRO) are also in preliminary design phases. It will be very important to have a fast neutron facility available in Europe to be able to perform irradiation experiments for fuels and materials and to develop and licence the advanced fuels. This will be one of the major roles of MYRRHA in particular.

In addition to Pu which will be recycled in fast reactors, a further reduction of high level waste radiotoxicity and the thermal power can be achieved by the extraction of the minor actinides Np, Am and Cm from the spent fuel and their transmutation in a fast neutron system. The reprocessing technologies and the fuel or target types have to be adapted to the selected option.

In all options, fuel steady state and transient experiments will need to be carried out to a large extend. It needs resources and is a long term challenge. That is the reason why in the demonstration projects now under design phase (ASTRID, MYRRHA), pure MOX fuel cores are taken as reference.

These irradiation infrastructures will then be used for testing the more advanced concepts.

4. EXAMPLES OF EURATOM FAST REACTOR PROJECTS

The Euratom Framework Programmes (cordis.europa.eu), the current FP7 2012–2013, and the future HORIZON 2020 from 2014 to 2018 support the fast reactor programme initiatives through several projects listed below:

— Fast reactor systems: CP-ESFR, JASMIN (Sodium fast ReacGoFastR, ALLIANCE (GFR), LEADER (LFR), EVOL (Molten Salt Fast Reactor);
— Safety: the EU SARGEN_IV project brings together the main European stakeholders to propose a European harmonized safety assessment practice for innovative reactors with fast neutron spectrum planned to be built in Europe;
— Fuel cycle safety: F-BRIDGE, FAIRFUELS, PELGRIMM (fuel behaviour), ACSEPT, SACSESS, ASGARD (fuel partitioning);
— Structural and cladding materials: GETMAT, MATTER
— Codes and basic data: NURISP, THINS (thermohydraulics and core physics), ANDES, EUFRAT (nuclear data);
— Transnational access to large infrastructures: TALISMAN (actinide sciences), ERINDA (nuclear data).

Each of these projects are co-financed by the European Commission and the participating Member States organizations. Since 2012, they are focusing on safety aspects upon the request of the European Council. The JRC is partner in many of the consortia implementing these projects.

Considering transmutation in ADS, the main EURATOM efforts are being placed on the safety assessment studies related to the MYRRHA demonstration projects by SCK/CEN in collaboration with European partners. Projects such as MAX, MAXSIMA, ARCAS and FREYA will include safety studies with a focus on transients potentially leading to pin failures.

In support and in addition to these European collaborative projects, the JRC carries out fast reactor and P&T research in its dedicated laboratories at Karlsruhe (Institute for Transuranium Elements), Geel (Institute for Reference Materials and Measurements) and Petten (Institute for Energy and Transport). The main JRC activities focus on reactor and fuel cycle safety studies and experiments, fuels and materials properties, and basic (actinides, nuclear data) supporting of research programmes. More particularly, the investigation of advanced fuel concepts and their safety relevant properties and irradiation behaviour, as well as separation strategies for closing the fuel cycle, are central elements of the work carried out at the JRC Institute for Transuranium Elements [10].

Spent fuel from existing and closed fast reactors represents an important knowledge legacy, opening the possibility for dedicated studies using techniques (e.g. thermal conductivity, isotope radial distribution, vaporization behaviour) not available 40 years ago when these fuels were licensed originally. Thermophysical (conductivity) and thermochemical (vaporization behaviour) property determination on irradiated and fresh fuels, as well as on model systems, needs to be expanded beyond today's database. Recently, in close collaboration with the French Commissariat à l'énergie atomique et aux énergies alternatives in the frame of the FBRIDGE project in the EU's Seventh Framework Programme, the JRC-ITU revisited the high temperature properties of plutonium dioxide and mixed oxide solid solutions using innovative approaches. For the melting studies, the potential of the fast laser heating technique developed at JRC-ITU [11] which creates self-crucible conditions was explored. For PuO_2, the experiments resulted in a value more than 300 degrees higher than the earlier values, 3017 ± 17 K versus 2673 K [12], a more than substantial difference. A slightly smaller difference was found for NpO_2 [13] and its melting temperature, 3070 ± 62 K, lies between PuO_2 and UO_2 (3130 ± 20 K). This leads to a new view on the trend of the melting temperatures of the actinide dioxides, as the nearly linear decrease from ThO_2 to PuO_2 suggested by previous results no longer applies. The newly established trend much better reflects the gradual changes in the interactions of f- and d-electron states in the light actinides. For the U-Pu-O ternary system, a fairly reliable understanding is now achieved. A major challenge will be to extend this system to americium, an important issue for fuels for the next generation of fast reactors, for which transmutation capabilities are foreseen.

In general, there is a push towards a greater fundamental understanding of fuels, whereby the breakthroughs in modelling and simulation at multiple time and distance scales are being reinforced with dedicated experiments. To take maximum advantage, a stronger coupling to experiment is required. One can envisage improved use of modelling for better designed experiments, geared at reducing the lengthy times and concomitant cost in fuel safety research. All such studies should lead to improved mechanistic and phenomenological models incorporated in engineering type codes accurately predicting in-pile behaviour.

5. R&D INFRASTRUCTURE

As highlighted in the SNETP Strategic Research and Innovation Agenda [14], besides the essential competences and experience of experts, the discussions above highlight the major investments needed in modern experimental infrastructure and facilities to enable the above R&D to be performed adequately. Fuel reference samples manufacturing laboratories able to handle transuranics, irradiation facilities for the investigation, characterization, development and, ultimately, validation and qualification of fuels, accident testing, generation of the appropriate and assessed data for code development and validation, and to increase the fundamental understanding of material and fuel behaviour are essential.

The availability of state of the art nuclear infrastructures and scientific facilities is essential to achieve the strategic objectives. In this context, the JRC also provides unique and complementary facilities. In fulfilling the mandate of the Euratom Treaty, it establishes basic installations necessary for the safe development of nuclear energy in the EU while promoting research and the dissemination of technical information. More recently, it strenghthened its policy in providing access to its unique scientific facilities to all Member States, thus contributing to complementary optimization of resources, both through FP7 Indirect Actions Networks of Excellence but also through JRC's institutional dedicated programmes.

The implementation of a systematic, rolling upgrade and renovation plan for the scientific infrastructures of the JRC was developed aiming at achieving the flexibility of operation necessary to cover key European applications within the overall perspective of the 'Sustainable Nuclear Energy Platform', and the related European Commission research programmes, and, at an international level, within the Generation IV International Forum (GIF) activities.

6. GENERATION IV INTERNATIONAL FORUM (GIF)

In 2006, Euratom, represented by the European Commission, has signed the GIF Framework Agreement, which organizes the joint efforts amongst its 10 members related to the R&D work on future reactor systems. Six of these systems are being considered, of which 4 are fast reactors (three cooled with sodium, lead or gas, and the fourth being a molten salt reactor). The JRC is the implementing agent coordinating Euratom participation in GIF. Euratom represents all European Member States (France being a direct member of GIF), and due to the various interests is also represented in each of the fast reactor systems. Euratom contributions to the projects in place are financed through three sources, namely (i) the indirect actions of the framework programmes as mentioned before, (ii) the direct research programme of the JRC, and (iii) the direct contributions of Member States organizations. The total effort of the broad Euratom contribution represents about 10% of the total yearly budget of the GIF projects. It should be noted that since 2012, the main emphasis has been put on reactor, materials and fuel cycle safety research, in response to the Fukushima accident.

7. TRAINING AND EDUCATION

Education and training in the field of nuclear science and technology is a key component of the nuclear infrastructure worldwide and of nuclear safety and security, as clearly stated at the G8 2009 Summit [15]. Concerns have been raised that nuclear education and training is not at the level it should be, as summarized in an OECD/NEA report [16]. These concerns, as well as the need to maintain the current high level of nuclear safety, led the Council of the European Union to conclude that it “is of the view that it is essential to maintain in the European Union a high level of training in the nuclear field” [17].

In response to this, universities in the EU and associated countries have initiated Master of Science degrees in nuclear science or nuclear technology or nuclear oriented specialization in other tracks during the last decades. They offer a wide scope of courses and training in the nuclear field for the MSc graduates, but they are generally lacking the possibility of specialization in tracks strongly related to the nuclear fuel cycle, for which the handling of nuclear materials is required. European universities have nowadays limited opportunities for working with radioactive materials in practical quantities.

Because of these specific infrastructure needs, nuclear education and training have been the common effort of universities and (inter)national laboratories in many countries. The role of the European Commission in this context was defined in the Euratom Treaty (Rome, 1957) in which it is explicitly mentioned that, “The Commission shall be responsible for promoting and facilitating nuclear research in the Member States and for complementing it by carrying out a Community research and training programme.” In the light of this, the JRC has taken the initiative to establish the European Nuclear Safety & Security School (EN3S) centred on the facilities of the JRC-ITU in order to to make its nuclear research facilities more accessible for training and education programmes.

ACKNOWLEDGEMENT

The author of this presentation wishes to thank all colleagues from the European Commission (Research and Innovation DG and the JRC in particular), and from European Member States organizations who are very actively involved in programming, financing and managing the research activities described here.

REFERENCES

[1] EUROPEAN COMMISSION, Energy Roadmap 2050, Communication (2011), http://ec.europa.eu/energy/publications/doc/2012_energy_roadmap_2050_en.pdf

[2] EUROPEAN ENVIRONMENT AGENCY, Greenhouse gas emission trends and projections in Europe 2011, EEA Rep. 4/2011, Copenhagen, Denmark (2011).

[3] EUROPEAN COMMISSION, The European Strategic Energy Technology Plan, http://ec.europa.eu/energy/technology/set_plan/set_plan_en.htm

[4] EUROPEAN COMMISSION, Proposal for a Council Regulation on the Research and Training Programme of the European Atomic Energy Community (2014–2018) complementing the Horizon 2020 – The Framework Programme for Research and Innovation; {SEC(2011) 1427}; {SEC(2011) 1428}, Communication (2011), http://ec.europa.eu/research/horizon2020/pdf/proposals/com(2011)_812_final.pdf

[5] MINISTRY OF FOREING AFFAIRS OF THE NETHERLANDS, Seoul Nuclear Security Summit, March 26-27, 2012, The Hague (2012), http://www.thenuclearsecuritysummit.org/eng_main/main.jsp

[6] EUROPEAN COMMISSION, Community Research and Development Information Service, Euratom Seventh Framework Programme (FP7), http://cordis.europa.eu/fp7/euratom/home_en.html

[7] EUROPEAN COMMISSION, Community Research and evelopment Information Service, Euratom Seventh Framework Programme (FP7), Indirect Actions, http://cordis.europa.eu/fp7/euratom/indirect_en.html

[8] EUROPEAN COMMISSION, Community Research and evelopment Information Service, Euratom Seventh Framework Programme (FP7), Direct Actions, http://cordis.europa.eu/fp7/euratom/direct_en.html

[9] EUROPEAN COMMISSION, SETIS Strategic Energy Technologies Information System, European Industrial Inititative on Sustainable Nuclear Energy, http://setis.ec.europa.eu/about-setis/technology-roadmap/european-industrial-initiative-on-sustainable-nuclear-energy

[10] FANGHÄNEL, T., et al., Safety of Advanced Nuclear Fuel cycles, these Proceedings.

[11] MANARA, D., et al., New techniques for high-temperature melting measurements in volatile refractory materials via laser surface heating, J. Rev. Sci. Instrum. **79** (2008) 113901.

[12] DE BRUYCKER, F., et al., J. Mater. Today **13** (2010) 52.

[13] BÖHLER, R., et al., J. Appl. Phys. **111** (2012) 113501.

[14] SUSTAINABLE NUCLEAR ENERGY TECHNOLOGY PLATFORM, Strategic Research Agenda 2012, Paris (2012), http://www.snetp.eu/www/snetp/images/stories/Docs-SRA2012/snetp_sria_for_public_consultation.pdf

[15] G8 NSSG Summit 2009, Nuclear Education and Training — Institutional Capacity-Building, L'Aquila, Italy (2009), http://www.g8italia2009.it/static/G8_Allegato/NSSG_ET-ICB_Final_28May_OK%5B1%5D,0.pdf

[16] OECD NUCLEAR ENERGY AGENCY, Nuclear Training and Eduction. Cause for Concern?, Summary Report, Paris (2000), http://www.nea.fr/ndd/reports/2000/nea2428-education.pdf
[17] COUNCIL OF THE EUROPEAN UNION, "I/A" Item Note, Brussels (2008), http://register.consilium.europa.eu/pdf/en/08/st15/st15406.en08.pdf

OECD NUCLEAR ENERGY AGENCY ACTIVITIES RELATED TO FAST REACTOR DEVELOPMENT

T. DUJARDIN, N.T. GULLIFORD
OECD Nuclear Energy Agency,
Paris

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 along two areas of activity: one focused on scientific research and technology development needs and one dedicated to strategic and policy issues. The paper summarizes recent and ongoing NEA activities in each of these areas of activity, including: improved evaluations of basic nuclear data needed for the development of fast reactor systems, expansion of integral experiments databases to provide improved validation for fast reactor modelling methods, modelling of transients in SFRs, creation of an innovative fuels expert group, a series of information exchange meetings on actinide and fission product partitioning and transmutation, study on homogeneous versus heterogeneous recycle of transuranic isotopes in fast reactors, studies on research needs and the availability of experimental facilities for fast reactor safety studies, and a study on trends towards sustainability in the nuclear fuel cycle. 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. The NEA will continue to support member countries in the field of fast reactor development and related advanced fuel cycles by providing a forum for exchange of information and various other collaborative activities.

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 (NSC) and the Nuclear Development Committee (NDC). This paper summarizes recent and ongoing NEA activities in the field of critical and non-critical fast neutron reactor system development.

2. NUCLEAR DATA AND VALIDATION OF MODELLING METHODS

2.1. Nuclear data needs for advanced reactors

Since 1989, the NEA has organized worldwide cooperation in the area of nuclear data evaluation. Currently this work is carried out under the auspices of the NEA NSC Working Party on International Nuclear Data Evaluation Co-operation (WPEC), which provides a framework for cooperative activities between the major nuclear data evaluation projects (e.g. ENDF, JEFF, JENDL, BROND and CENDL) with the objective of improving the quality and completeness of evaluated data libraries available for use in science and technology. The work within the Working Party is organized into subgroups, one subgroup for each study.

In 2008, WPEC released the conclusions of a comprehensive study focused on the development of a systematic approach to define data needs for advanced reactor systems. The final report describes sensitivity and uncertainty analyses of the most significant integral parameters related to the core and fuel cycle, as well as a target accuracy assessment complemented by quantitative evaluations of nuclear data improvement required to meet those target accuracies.

This WPEC study was followed by two complementary works that are being finalized this year. The first one was established to review the accuracy of current experimental and evaluated nuclear data required for advanced reactor systems and to provide recommendations for new nuclear data experiments and/or evaluations to meet target system accuracies. The second one was established to study methods and issues relative to the assimilation of accurate integral experiment information in the nuclear data adjustment process with the same objective of meeting target system accuracies.

In addition, WPEC recently investigated calculated-over-experiment ratio (C/E) discrepancies in fast uranium-core integral parameters observed with all major evaluated libraries. The final report reviews the ^{235}U capture cross-section data and describes sensitivity analyses performed to better understand the influence of other nuclear data on integral parameters. This study shows a possible overestimation of the ^{235}U capture cross-section in the 0.1–2.5 keV energy range and provides recommendations for future work.

2.2. Expansion of integral experiment databases to provide improved validation for fast reactor modelling methods

2.2.1. NEA databases of integral experiments

As part of its objectives to help build the scientific and technical basis for the development of future generation nuclear systems and to preserve essential knowledge in the field of nuclear science, an extensive programme of work to evaluate data from integral experiments has been established, including reactor physics, shielding and criticality safety experiments on fast reactor systems. Of these NEA database projects, two in particular contain information applicable to the validation of neutronics modelling methods for fast reactor studies, namely, the International Reactor Physics Evaluation Project (IRPhEP) and the International Criticality Safety Benchmark Evaluation Project (ICSBEP).

Data from experimental facilities are reviewed and, if necessary, archives of information are made safe. This may typically involve the indexing and scanning of key documents and archiving of logbooks, for example. Selected experiments go through a detailed evaluation process and where deemed appropriate, a benchmark description is created in a standardized format for inclusion in one of the international databases.

The details of the status of these activities is given in another paper at this conference along with the status of recent preservation activities for fast reactor archives in the United Kingdom.

2.2.2. Integral experiments for minor actinide management

The establishment of a reliable and economical fuel cycle, including the 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 is crucial for the detailed design of the transmutation systems, as well as for the accurate prediction of the spent fuel composition. Compared to the major actinides, integral experiments for the minor actinides are scarce due to the restrictions and difficulties in the material handling, sample preparation and post-treatment technology.

The NEA therefore launched a study 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 promoting integral experiments for minor actinide management. This work is nearing completion and a final report will be published later this year.

3. REACTOR SYSTEMS, MATERIALS AND THE FUEL CYCLE

3.1. Sodium cooled fast reactors (SFRs)

With respect to the safety performance of sodium cooled fast reactors (SFRs), one of the foremost objectives of the Generation-IV International Forum (GIF) 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.

An NEA task force has been created to address the above mentioned objective by performing a comparative analysis of the safety characteristics of two different core sizes: a large size core (3600 MW(th)) and a medium size core (1000 MW(th)). For both core 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, discussion and agreement on corresponding calculation methodology. The study is expected to be completed in 2013.

3.2. Heavy liquid metal technology

Lead and lead–bismuth eutectic alloys are chemically inert, have a high boiling temperature and 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. The purpose was to develop standards, identify areas where further studies are needed and help establish a common methodology for experiments and data analysis. This handbook is currently being reviewed and updated and version 1 of the handbook will be published early 2014. The updated version will include the most recent developments (since 2007). Nine institutes and national laboratories are contributing to this review. The structure of the new version will be similar to the first one and contain four chapters summarizing key aspects of thermohydraulics, instrumentation and systems technologies. The last three chapters will 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 (LACANES). This benchmark focuses mainly on characterizing the thermal hydraulic behaviour of such systems under the steady-state forced or under natural convection, which is of critical importance for system design development. The first phase of the characterization of the system and study of the steady-state forced case has now been completed [6]. Understandable guidelines for prediction pressure loss were obtained based on comparison between many predictions calculated by handbook correlations or CFD simulations. From these activities, a better understanding of pressure loss modelling in lead alloy cooled systems was obtained. The LACANES benchmark Phase-II in the case of natural circulation is still ongoing.

3.3. Innovative fuels

The dawn of advanced nuclear systems requires further insights into the new types of fuel and materials that are currently being developed for use in these systems. An understanding of the technical issues associated with the implementation of innovative fuels (ADS fuels, oxide, metal, nitride, carbide fuels) and clad materials for use in advanced fuel cycle is necessary.

To gain further understanding, the NEA has launched a study on these fuels, especially those that contain minor actinides as opposed to standard fuels. A standard report was prepared by the Expert Group on Innovative Fuels and will be published later this year.

3.4. Advanced fuel cycle scenarios

The implementation of fast reactors and associated fuel cycles requires an understanding of the scientific and economic issues associated with the transition for current fuel cycles.

The NEA Expert Group on Fuel Cycle Transition Scenarios was created in 2003 to study R&D needs and relevant technology for an efficient transition from current to future advanced reactor fuel cycles. After reviewing national, regional or worldwide transition scenarios, the expert group performed a benchmark study to compare the existing codes in terms of capabilities, modelling and results. The benchmark was conducted in two phases: (1) depletion calculations for PWR UOX, PWR MOX and fast reactor calculations and (2) transition calculation using various scenario codes (using three different transition scenarios: once-through, limited plutonium recycling in LWRs and plutonium and minor actinides recycling in fast reactors.

3.5. Recycling technologies in fast reactors

There has been international interest in technologies and strategies to be deployed to minimize wastes and a wide consensus has emerged to prove the advantages of the approach based on fast reactors. The existing technologies and significant operational experience gained in the treatment of spent fuels from thermal power reactors are a good starting point for new recycling technologies but need to be adjusted to the specifics of fast reactor fuels. Fuel transuranics recycling will become a mandatory feature to pursue resource sustainability and waste minimization.

The NEA Expert Group on Fuel Recycling Chemistry (formerly Chemical Partitioning) is focusing on recycling technologies for spent nuclear fuel including waste treatment. Technical assessments of separation processes are being prepared in applications related to the current and future nuclear fuel cycles and recommend collaborative international efforts to further processes development. The current study involves the assessment of separation processes, in particular separation of minor actinides and for different fuel cycle scenarios (including 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 un-separated 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 recycling in fast reactors could be to separate out the less radioactive component of the spent nuclear fuel, e.g. plutonium or combined neptunium with plutonium, 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 actinides target fuels is called heterogeneous recycling.

Potential advantages and disadvantages of both homogeneous and heterogeneous recycling modes have been defined and investigated in 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 diverse perspectives and are based on different objectives, hypotheses and experimental results.

A recently completed NEA study compares the criteria for a choice 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 assesses the potential impact, both on the reactor core and on the power plant. Fuel and target related issues are 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 reflects previous, and ongoing or planned irradiation programmes. Specific scenario studies have been considered in order to underline specific needs and requirements, both for the short and long terms. The final report is scheduled for publication later this year.

3.6. Partitioning and transmutation (P&T)

Alternative technologies to decrease the amount of high level radioactive waste to be stored in deep geological repositories are being developed, such as P&T using either critical or subcritical fast reactor systems. The development of P&T technology has the potential of minor actinides separation using either aqueous and pyroprocessing, demonstrating high separation factors at laboratory scale.

In order to give experts a forum to present and discuss state-of-the-art developments in the P&T field, the NEA has been organizing biennial information exchange meetings on actinide and fission product partitioning and transmutation since 1990. The meetings covered scientific, as well as strategic/policy developments in the field of P&T, such as fuel cycle strategies and transition scenarios, radioactive waste management strategies (including secondary wastes), progress in fuels and materials, related physics and experiments, system design and economics of P&T.

The last NEA Information Meeting on Actinide and Fission Product Partitioning and Transmutation was held in the Czech Republic in September 2012 where a total of 90 papers were presented. The 13th meeting will be held in September 2014 and the Republic of Korea has offered to host.

3.7. Materials

NEA activities on materials for fast reactor systems aim at providing the necessary support to the development of innovative fuels and structural materials needed for the development of new technologies with fast spectrum and different coolants (liquid metal or helium). A large spectrum of subjects is covered, from fundamental R&D on modelling methods to reviews on innovative fuels and structural materials. The NEA also supports the organization of workshops in the field of innovative material research.

More specifically, these activities cover:

- Modelling methods: the Working Party on Multi-scale Modelling of Materials (WPMM) was established in 2009 with the aim of providing an assessment of the possibilities and limits of numerical methods applied to multi-scale modelling, with a specific focus on the applications to future development of fuels, the modelling of irradiations in structural nuclear materials and an attempt to validate these methods with appropriate benchmarks.
- Innovative fuels and structural materials: two Expert Groups in the Working Party on the Scientific Issues of the Fuel Cycle (WPFC) accomplished, in 2012, a review on the current status of innovative fuels and structural materials. In terms of fuels, all solutions from oxide, to carbide, nitride, metal, dispersion fuels and special mechanical fuels (sphere-pac, vibro-pac) were considered, including an evaluation of the technical readiness level for fabrication and irradiation performances. The Expert Group on Innovative Materials carried out an analogue review on innovative cladding and structural materials being developed for SFRs, lead cooled fast reactors (including accelerator driven systems) and gas cooled fast reactors (GCRs).
- TAF-ID: This project on Thermodynamic Data of Advanced Fuels — International Database, externally funded by 9 organizations and 7 countries, started at the beginning of 2013 with the aim of creating a database of thermodynamic data for advanced fuels, needed for a large spectrum of applications, from P&T in FBR to safety analysis (fuel cladding interaction, corium behaviour, etc.).
- SMINS: the second edition of the International Workshop on Structural Materials for Innovative Nuclear Systems (SMINS) was held in 2010 in Daejon (Republic of Korea) and the third edition will be hosted by Idaho National Laboratory at Idaho Falls (United States of America) in October 2013. This workshop promotes exchange of information on structural materials R&D, covering fundamental studies, advanced modelling and experiments.

4. STRATEGIC ISSUES

4.1. Trends in nuclear fuel cycle

Increased interest in expanding nuclear power to cope with rising demand for energy and potential climate change impacts puts increased focus on the nuclear fuel cycle and whether there are significant moves towards ensuring sustainability for the long term. Future nuclear power programme decisions will be increasingly based on strategic considerations involving the complete nuclear fuel cycle, as illustrated by the joint projects for Generation-IV reactors. The recently published study "Trends towards Sustainability in the Nuclear Fuel Cycle" (NEA, 2011) reviews developments over the last ten years and potential changes, in the near and longer terms, in technologies and government actions (both national and international) related to the fuel cycle, with a specific focus on sustainability. The report shows that although major breakthroughs in technology have not occurred in the recent past and are not expected in the near future, the nuclear sector has seen a continuous evolution, driven mostly by the industry, with incremental changes in mainstream reactors and fuel cycle technologies aimed at their optimization. As projections have not indicated immediate constraints from shortage of resources, there has been little incentive for significant investment in advanced fuel cycles and/or in closing the fuel cycle. Nevertheless, it has become clear that step changes in sustainability are linked to the development and uptake of advanced fuel cycle technologies, which, however, require national energy policy changes and longer term strategic choices and cannot be driven purely by market forces. The commercial deployment of Generation-IV nuclear reactors is an important step in this context; their development aims at enhancing the safety, economics, sustainability, proliferation resistance and physical protection of future nuclear systems, while holding the promise of opening nuclear applications beyond today's electricity production.

4.2. Fuel cycle transition scenarios

Through greater recycling of spent fuel, advanced fuel cycles would allow a better use of the energy potential of natural uranium. In particular, closed fuel cycle schemes based on fast reactors could greatly decrease the requirement for fresh uranium. At the same time, the volume and radiotoxicity of waste could be decreased significantly, through a selective separation (partitioning) of the long lived elements, including minor actinides (e.g. neptunium, americium and curium) and possibly some fission products from spent fuel. Mixed directly with the fuel (homogeneous transmutation) or incorporated in separate targets (heterogeneous transmutation), these separated isotopes could then be transformed into shorter lived elements (transmutation) by fissioning or neutron capture either in reactors or in specifically designed systems (e.g. ADS). Alternatively, the partitioned isotopes could be vitrified as waste in special matrices and separately conditioned and disposed of.

The deployment of these advanced cycles, including Generation-IV systems and P&T technologies, requires significant R&D advances, with some important challenges still lying ahead. Some of these are listed below:

- R&D on novel fuel separation methods, including processing techniques, for waste streams that can be destined to disposal, or re-processing followed by transmutation;
- Advances in transmutation technologies;
- Addressing emerging waste management issues linked to new systems and advanced separation technologies, such as the development of new conditioning processes, a better characterization and optimization of composition and quantities of waste streams (including LILW).

4.3. Experimental facilities for fast reactor safety studies

For many years, the NEA has been addressing advanced reactors and developing related information useful to regulators, designers and researchers on safety issues and research needed. In particular, the NEA Committee on the Safety of the Nuclear Installations (CSNI) and the Committee on Nuclear Regulatory Activities (CNRA) organized a joint workshop on the Role of Research in a Regulatory Context (RRRC-2, 5 December 2007). Among other topics, the workshop addressed the challenges that the nuclear community will face when performing safety evaluations of advanced reactor designs, the research that may be needed to perform the reviews, and the possible means for jointly conducting this research. In particular, the workshop discussed research topics relevant for GCRs and SFRs and recommended that CSNI organizes a task group to identify the needed research and recommend a path forward.

The CSNI Task Group on Advanced Reactor Experimental Facilities (TAREF) was initiated in 2008 based on these discussions and was aimed at providing an overview of facilities suitable for carrying out the safety research that was considered necessary for GCRs and SFRs. Other reactor systems could be considered in a subsequent phase. Although GCRs are relevant for this paper, only SFRs are addressed in the following paragraphs.

As a general approach, the TAREF members decided to build on the experience of a similar activity conducted by CSNI which focused on facilities suitable for current and advanced water reactor systems. In particular, the SFEAR method was adopted for both designs, which entailed first identifying high priority safety issues that require research, and then categorizing the available facilities in terms of their ability to address the safety issues. Based on this survey and consistent with the SFEAR report, the technical areas to be addressed for SFRs were identified as being: (a) thermo-fluids; (b) fuel safety;

(c) reactor physics; (d) severe accidents; (e) sodium risks; (f) structural integrity; (g) other issues.

For each of the above technical areas (a to g), the TAREF members agreed on a set of safety issues needing research and established a ranking with regard to safety relevance (high, medium, low) and status of knowledge based on the following scale relative to full knowledge: high (100–75%), medium (75–25%), low (25–0%). Similar to GCR issues, only the issues identified as being of high safety relevance and for which the state of knowledge is low or medium were included in the discussion, as these issues would likely warrant further study.

For each of the safety issues, the TAREF members identified the related facilities that were deemed appropriate to address the issues in question, providing relevant information such as operating conditions (in- or out-of-reactor), operating range, description of the test section, type of testing, instrumentation, current status and availability, uniqueness, etc.

Based on the information that was assembled on both safety issues and related facilities, the TAREF members assessed prospects and priorities for SFR safety research and developed recommendations as to priorities and options regarding facility utilization through programmes that could be pursued internationally.

The group members agreed that for new SFR projects, the most important and top tier R&D safety needs concern the technical areas with the following priority order: fuel safety (b) and severe accident (d) issues are of prime interest due to the lack of knowledge on new pin design and materials, while thermo-fluids (a) and core physics issues are of second priority as one can live with the current knowledge when considering some margins to cover uncertainties. Sodium risks (e) and structural integrity (f) issues may be considered with third priority as they are more design dependent.

The need for fuel pin irradiation capabilities under representative conditions of fast neutron flux has been identified as a crucial point for addressing high priority safety issues.

As a result of the TAREF activity, an OECD joint project was proposed by the Japan Atomic Energy Agency (JAEA) and was set up in the JAEA High Temperature Engineering Test Reactor (HTTR). The objectives of this project are to conduct integrated large scale test of loss-of-forced cooling in the JAEA HTTR reactor to examine high temperature gas cooled reactor (HTGR) safety characteristics in support of regulatory activities and to provide data useful for code validation and improvement of simulation accuracy. The reactor performance in the accidental condition considered in the Phenomena Identification Ranking Tables set up by the USNRC will be assessed in this project.

It is expected that other OECD joint projects will be initiated based on the recommendations issued from the TAREF activity. In particular, OECD joint projects addressing first priority SFR safety issues might be initiated within two or three years.

5. CONCLUSION

While there has been a shift in emphasis following the Fukushima accident to research in the context of the existing fleet of nuclear power plant, it is also true that there remains a high level of interest in the continued development of advanced nuclear systems and fuel cycles. Through greater recycling of spent fuel, advanced fuel cycles would allow a better use of the energy potential of natural uranium. In particular, closed fuel cycle schemes based on fast reactors could greatly decrease the requirement for fresh uranium

Ambitious R&D programmes have been undertaken at the national level in many countries and in the framework of several international projects; they are expected to lead to the design and development of advanced reactors and fuel cycle facilities responding to the sustainable development goals of governments and society. The NEA will continue to support member countries in the field of fast reactor development and related advanced fuel cycles by providing a forum for exchange of information and various other collaborative activities.

BIBLIOGRAPHY

OECD NUCLEAR ENERGY AGENCY

Trends in the Nuclear Fuel Cycle: Economic, Environmental and Social Aspects, OECD Publishing, Paris (2001).

Nuclear Safety Research in OECD Countries: Support Facilities for Existing and Advanced Reactors (SFEAR), OECD Publishing, Paris (2007).

Handbook on Lead-bismuth Eutectic Alloy and Lead Properties, Materials Compatibility, Thermalhydraulics and Technologies, OECD Publishing, Paris (2007).

Uncertainty and Target Accuracy Assessment for Innovative Systems Using Recent Covariance Data Evaluation, International Evaluation Co-operation, NEA/WPEC-26, Vol. 26, OECD Publishing, Paris (2008).

International Evaluation Co-operation, Volume 26, Uncertainty and Target Accuracy Assessment for Innovative Systems using recent Covariance Data Evaluations, Paris (2008), http://www.nea.fr/html/science/wpec/volume26/volume26.pdf

"Structural Materials for Innovative Nuclear Systems (SMINS)" Proc. Workshop, OECD Publishing, Paris (2008).

"Utilisation and Reliability of High Power Proton Accelerators (HPPA5) ", Proc. Workshop, OECD Publishing, Paris (2008).

Fifth Workshop on Advanced Reactors with Innovative Fuels (ARWIF), Proc. Workshop, OECD Publishing, Paris (2009).

Nuclear Fuel Cycle Transition Scenario Studies, OECD Publishing, Paris (2009).

Tenth Information Exchange Meeting on Actinide and Fission Product Partitioning and Transmutation", Proc.Workshop, OECD Publishing, Paris (2009).

Nuclear Fuel Cycle Synergies and Regional Scenarios for Europe, OECD Publishing, Paris (2009).

Strategic and Policy Issues Raised by the Transition from Thermal to Fast Nuclear Systems, OECD Publishing, Paris (2009).

Experimental Facilities for Gas-cooled Reactor Safety Studies, OECD Publishing, Paris (2009).

Proc. Int. Workshop Structural Materials for Innovative Nuclear Systems, Daejeon, Korea (2010) http://www.nea.fr/html/science/smins2/

WPFC Task Force on Homogenous versus Heterogeneous Recycle of Minor Actinides in Fast Reactors (WPFC/TFHH), (2010)
http://www.nea.fr/html/science/wpfc/index_hh.html

First International Workshop on Technology and Components of Accelerator-driven Systems, Karlsruhe, Germany (2010)
http://www.nea.fr/html/science/wpfc/tcads/1st/

Experimental Facilities for Sodium Fast Reactor Safety Studies, OECD Publishing, Paris (2011).

Uranium-235 Capture Cross-Section in the keV to MeV Energy Region, Int. Evaluation Co-operation, NEA/WPEC-29, Vol. 29, OECD Publishing, Paris (2011).

Trends towards Sustainability in the Nuclear Fuel Cycle, OECD Publishing, Paris (2011).

Benchmark on Thermal-hydraulic Safety Issues of Lead Alloy-cooled Advanced Nuclear Energy Systems, Phase I: Isothermal forced convection case, OECD Publishing, Paris (2012).

Benchmark Study on Nuclear Fuel Cycle Transition Scenarios Analysis Codes, OECD Publishing, Paris (2012).

Expert Group on Integral Experiments for Minor Actinide Management (2012)
http://www.nea.fr/html/science/ma/

Homogeneous vs Heterogeneous Recycle of Transuranics in Fast Nuclear Reactors, OECD Publishing, Paris (2013).

Activities in Preserving, Evaluating and Applying Data from Fast Reactor Experiments, Proc. of FR-13, OECD Publishing, Paris (2013).

WPFC Expert Group on Innovative Structural Materials (WPFC/ISM) (2013)
http://www.nea.fr/html/science/wpfc/index_ism.html

Expert Group on Structural Materials Modelling (SMM), (2013)
http://www.nea.fr/html/science/wpmm/expert_groups/smm.html

WPFC Task Force on Benchmarking of Thermal-Hydraulic Loop Models for Lead Alloy-ooled Advanced Nuclear Energy Systems (LACANES), (2013)
http://www.nea.fr/html/science/wpfc/index_lacanes.html

Meeting Nuclear Data Needs for Advanced Reactor Systems, International Evaluation Co-operation, NEA/WPEC-31, Vol. 31, OECD Publishing, Paris (in press).

Methods and Issues for the Combined Use of Integral Experiments and Covariance Data, International Evaluation Co-operation, NEA/WPEC-33, Vol. 33, OECD Publishing, Paris (in press).

STATE-OF-THE-ART, NEW TRENDS AND DEVELOPMENTS IN THE FIELD OF FAST REACTORS AND RELATED FUEL CYCLES*

F. CARRE
Commissariat à l’énergie atomique et aux énergies alternatives,
France

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

FAST REACTOR DESIGNS: GOALS AND PATHS OF PROGRESS

Chairpersons

J.C. GARNIER
France

Y. KIM
Republic of Korea

STATUS OF SFR DEVELOPMENT IN THE REPUBLIC OF KOREA

YEONG-IL KIM, JINWOOK CHANG,
YONG-BUM LEE, CHAN-BOCK LEE, DOHEE HAHN
Korea Atomic Energy Research Institute,
Daejeon, Republic of Korea

Abstract

To reduce the volume and radiotoxicity of high level waste for final disposal, sodium cooled fast reactor (SFR) technologies are being developed in the Republic of Korea. The final goal of the SFR development effort is the construction of a prototype SFR by 2028 with an intermediate goal of obtaining design approval from the regulatory authority by 2020. Based on the experiences gained from the KALIMER conceptual designs, a conceptual design for the prototype SFR has been under development from 2012. To support the design work, various R&D activities, such as construction of a large sodium thermohydraulic test facility, the development of an under-sodium viewing technique, and metal fuel and computer code development and validation, are also performed.

1. SFR DEVELOPMENT PLAN

The sodium cooled fast reactor (SFR) technology development efforts in the Republic of Korea commenced in June 1997 with the approval of the Korean Atomic Energy Commission (KAEC) for a national mid- and long term nuclear R&D programme. The efforts have been focused on the development of key SFR technologies

A long term development plan for a demonstration SFR was authorized to provide a consistent direction to long term R&D activities by KAEC in December, 2008. This long term plan has been implemented under the nuclear R&D programmes of the National Research Foundation with financial support from the Ministry of Education Science and Technology. The plan was updated by KAEPC (Korea Atomic Energy Promotion Council) in November 2011 to be more realistic, reflecting the maturity of the technology achieved and the budget condition. The updated plan includes the design development of a prototype SFR by 2017, its design approval and construction by 2020 and 2028, respectively, as shown in Fig. 1. According to the plan, a conceptual design of the prototype SFR is being developed from 2012.

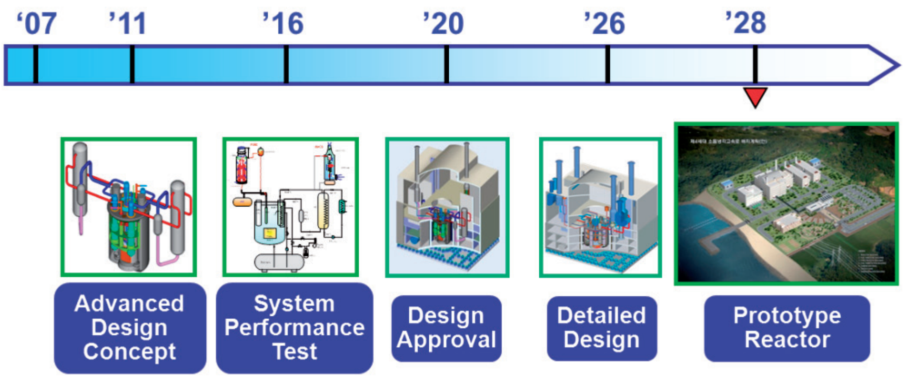


FIG. 1. SFR development plan.

To perform the prototype SFR development efficiently, the SFR Development Agency (SFRA) was organized as of 16th May 2012. The role of the SFRA is to secure the budget and manage the SFR development project.

2. CONCEPTUAL DESIGN OF A PROTOTYPE SFR

2.1. Top-tier design requirements

The objectives of the prototype SFR are to test the performance of metal fuels containing TRU (transuranics), which will be used later for a commercial SFR, and to demonstrate the TRU transmutation capability of a burner reactor. The prototype SFR will demonstrate the transmutation of TRUs recovered from the PWR spent fuel, and hence the benefits of the integral recycling of actinides in a closed fuel cycle in nuclear waste management.

Based on the objectives above, the top-tier design requirements for the prototype SFR and related design parameters that were extensively discussed to support their final determination are given in Table 1. The lessons learned from fast reactor programmes and the operating experiences of fast reactors worldwide, as well as the experiences gained from the development of KALIMER conceptual designs [1, 2], have been incorporated in the top-tier design requirements.

The capacity of 150 MW(e) was determined by the level of developed technology, investment risk, economics, and performance for TRU transmutation. The trade-off study on power level concluded that the power level should be at least 150 MW(e) to achieve the objectives of the prototype SFR.

TABLE 1. SUMMARY OF TOP-TIER DESIGN REQUIREMENTS FOR PROTOTYPE SFR

General design requirements	Reactor type: pool type Plant size: 150 MW(e) Plant design lifetime: 60 years Design basis earthquakes (SSE: 0.3g) Initial core: U–Zr metal fuel Reference core: U–TRU–Zr metal fuel
Safety and investment protection	Design simplification Negative power reactivity coefficient Core damage frequency: <10^{-6}/reactor-year No fuel cladding liquid phase propagation during DBEs Diversified core shutdown mechanism Reliable and diversified decay heat removal Accommodating unprotected ATWS events without any operator's action Large radioactivity release: <10^{-7}/reactor-year 3 days grace time without any operator's action for DBEs
Performance and economy	Plant availability: ≥70% Refueling interval: ≥6 months 100% off-site load rejection without a plant trip

2.2. Core design

A conceptual core has been designed for a prototype SFR with a 150 MW(e) power level. It is scheduled to use uranium fuel for the early stage of the prototype SFR operation since the database for the TRU-bearing metal fuel will not be sufficient for the period. Lead test rods and subassemblies of TRUs will be irradiated in this prototype reactor core to accumulate the TRU database. By doing so, the core will finally move to a TRU core.

As the first step, a favourable uranium core was searched to satisfy following design criteria:

(1) The fuel will be a U–Zr alloy and have less than 19.5 wt% enrichment.

(2) The cladding material will be D9 austenitic stainless steel to secure the mechanical integrity under the core exit temperature of 545°C.

(3) The bundle pressure drop without uncertainty will be less than 0.26 MPa.
(4) The fast neutron fluence will be less than 2.0×10^{23} n/cm^2 and the damage of D9 cladding by neutron irradiation will be less than 100 dpa.
(5) The cladding mid-wall temperature will be less than 650°C to ensure the limitation of cladding cumulative damage fraction (CDF) of less than 0.001.

Figure 2 and Table 2 show the core configuration and the performance of a candidate U core optimized by parameter studies. As shown in these results, all design criteria were fulfilled by adopting two enrichment zoned cores, especially to reduce the peak fast neutron fluence. To secure the safety characteristics from a neutronics perspective, several reactivity coefficients were evaluated, as shown in Table 3, and all negative reactivity feedbacks were confirmed except the sodium density coefficient at EOEC. Based on these parameters, a shutdown margin was also determined, and revealed a sufficient margin of –7.40 and 3.03 by primary control rods and secondary rods, respectively.

2.3. Fluid system design

The design concepts of the fluid system for a 150 MW(e) prototypical SFR are being developed. From the trade-off study to enhance plant safety and improve the economics and performance, the system concepts were established, and the design requirements and size of the main components were set up based on proven technologies [3]. The fluid system consists of a heat transport system including a primary heat transport system (PHTS), intermediate heat transport system (IHTS), and power conversion system (PCS), and a safety design feature such as a decay heat removal system (DHRS), as shown in Fig. 3. Also, the plant heat balance at 100% normal operating condition is shown in Fig. 4.

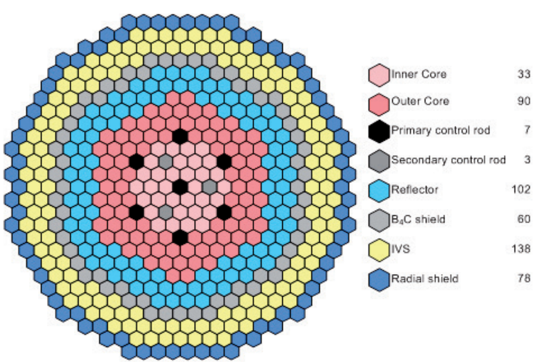


FIG. 2. Uranium core configuration of a 150 MWe prototype SFR.

TABLE 2. KEY DESIGN PARAMETERS (EQUILIBRIUM CYCLE OF URANIUM CORE)

Parameter		Value
Core electric power (MWe)		150
Core thermal power (MWt)		392.6
Core mixed mean inlet/outlet temp. (?)		390/545
Total flow rate (kg/s)		1991.8
Effective full power day (EFPD)		290
Number of batches (inner core/outer core)		5/5
Number of fuel assembly types		2
Active core height (mm)		1050
Enrichment (IC/OC) (wt.%)		13.0/19.5
Reactivity swing (pcm)		1160
Burnup (MWD/kg)	Avg.	49.0
	Peak	79.2
Fast neutron flux ($\times 10^{15}$ n/cm$^2\cdot$sec)	Avg.	0.95
	Peak	1.50
Peak fast neutron fluence ($\times 10^{23}$ n/cm^2)		1.90/1.90
Linear power density (W/cm)	Avg.	102.2
	Peak	181.6
Avg. power density (W/cm^3)		136.6
Pressure drop (MPa)		0.255

TABLE 3. REACTIVITY COEFFICIENTS (EQUILIBRIUM CYCLE OF URANIUM CORE)

	BOEC	EOEC
Doppler coeff. ($\Delta\rho/\Delta T$, pcm)	$-11163.8 \times T^{-1.442}$	$-11007.2 \times T^{-1.438}$
Axial expansion coeff. (pcm/ºC)	–0.17436	–0.17879
Radial expansion coeff. (pcm/ºC)	–0.55154	–0.56369
Sodium void effect (pcm)	–139.2	–41.3
Sodium density coeff. (pcm/ºC)	–0.02842	0.00415
Control rod worth (pcm)		
Primary	9781	9993
Secondary	4119	4238
Total	13 320	13 653

The PHTS is a pool type system in which all the primary components and primary sodium are located within a reactor vessel. Two mechanical PHTS pumps and four intermediate heat exchangers (IHXs) are immersed in the primary sodium pool. Each IHX has a thermal capacity of 98.2 MW(t) and is a counter flow shell and tube type with a vertical orientation inside the reactor vessel, where PHTS sodium flows downward through the shell side and IHTS sodium flows upward through the tube side. It transfers the heat generated in the core to the IHTS. The PHTS pump is a centrifugal type mechanical pump with a capacity of 69.6 m^3/min, and its inlet and outlet are located in the cold sodium pool. The core inlet and outlet temperatures are 390°C and 545°C, respectively.

The IHTS has two loops and there are two IHXs connected to one steam generator and one IHTS pump in each loop. Each IHTS pump is a centrifugal type with a capacity of 53.0 m^3/min and is located in each cold leg of the two IHTS loops. Each steam generator has a thermal capacity of 196.8 MW(t). The IHTS sodium flows downward through the shell side while the water/steam goes up through the tube side. The cold leg of the IHTS piping is a bottom up U-shaped unit with sufficient height to prevent sodium–water reaction products from reaching the IHX in the case of a steam generator tube failure.

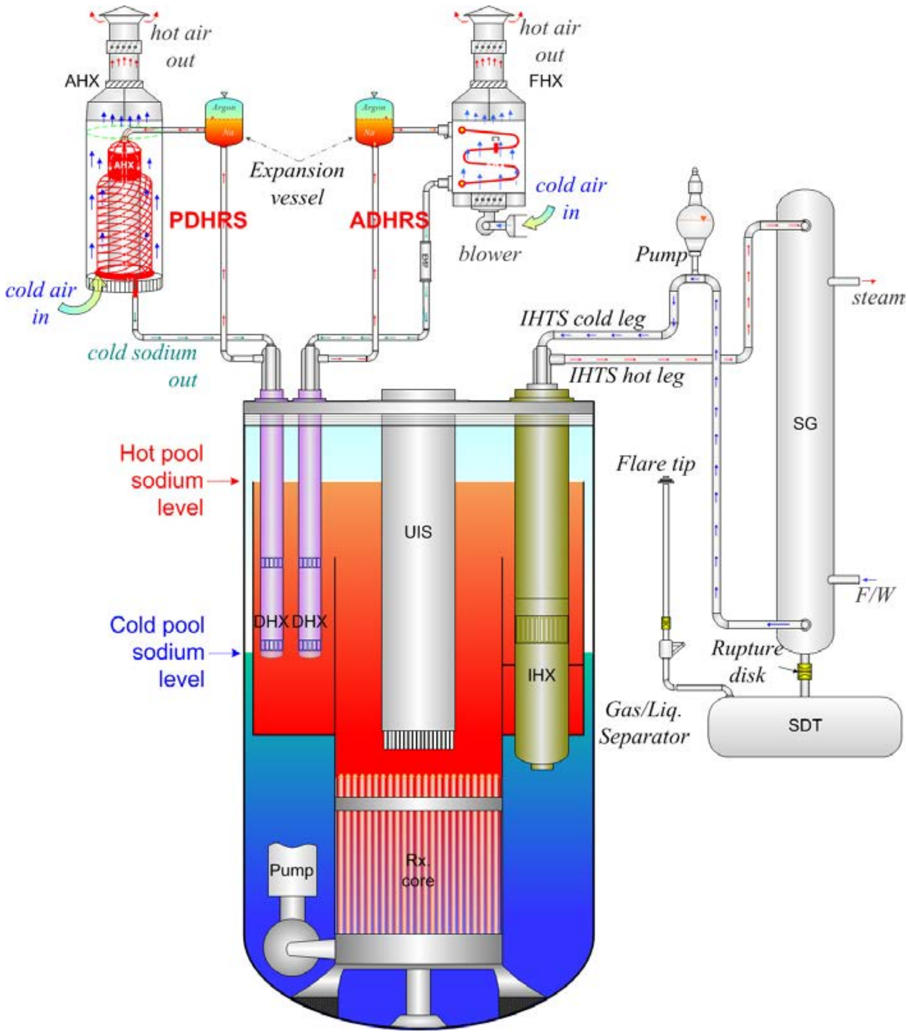


FIG. 3. Schematic of fluid system.

The DHRS, one of the safety design features, is composed of two passive decay heat removal systems (PDHRS) and two active decay heat removal systems (ADHRS). It is used to remove the decay heat of the reactor core after a reactor shutdown when the normal heat transport path is unavailable. It is a safety-grade system and is designed to have a sufficient capacity to remove the decay heat in all design basis events by incorporating the principles of redundancy and independency.

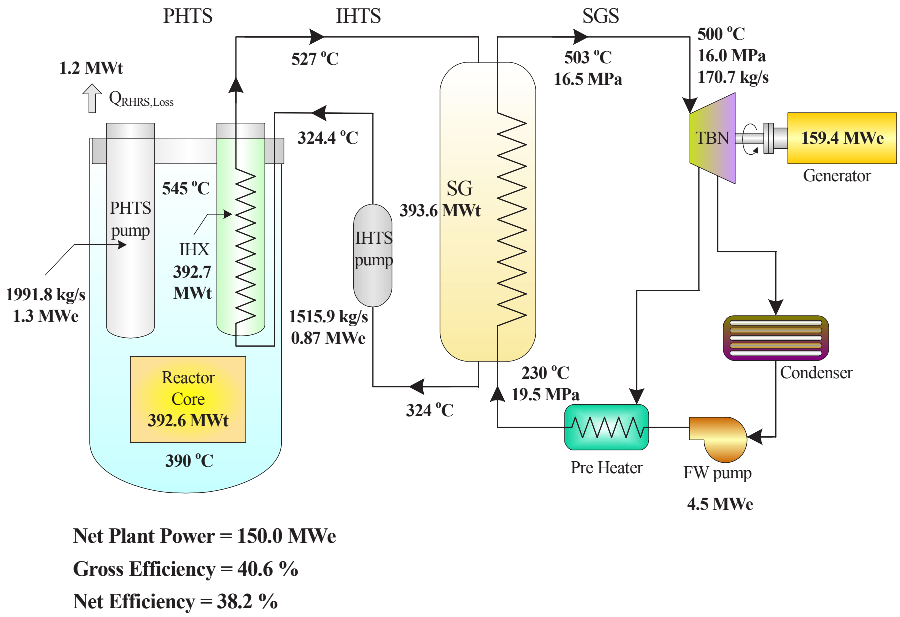


FIG. 4. Operating conditions at 100% plant power.

Each PDHRS is comprised of a decay heat exchanger (DHX) immersed in the reactor hot pool and a natural draft sodium-to-air heat exchanger (AHX) located in the upper region of the reactor building. It is operated by the natural circulation induced by the density and the elevation difference between the DHX and AHX.

Each ADHRS consists of a DHX, a forced-draft sodium-to-air heat exchanger (FDHX), an electromagnetic pump, and an FDHX blower. The electromagnetic pump and FDHX blower derive the sodium circulation in the loop and the air flow in the shell side of the FDHX, respectively. Because the ADHRS can also be operated in a natural convection mode against a loss of power supply, the heat transferred to the DHRS can be finally dissipated to the atmosphere through the AHXs and FDHXs by the natural convection mechanism of sodium and air only.

2.4. Mechanical structure concept design

The mechanical structure design of SSCs (structures, systems and components) is carried out in the conceptual level stage. The main mechanical design goals will be achieved by a simple reactor enclosure system, advanced design technologies and advanced design materials.

For a simple reactor design, the reactor vessel has a uniform thickness and there are no penetrations or attachments to the reactor vessel. The mechanical structure design of SSCs will be carried in compliance with the elevated temperature design rules of ASME BPV III, division 5. The horizontal seismic isolation design will be adapted for a reactor island, including a reactor building, an auxiliary building and a wastage/maintenance building. For the design materials, 9Cr-1Mo-V steel is used for the IHX, DHX, IHTS piping, and steam generator. Others, such as the reactor vessel, reactor internal structures and reactor head, are composed of 316 stainless steel.

Figure 5 shows the conceptually designed PHTS arrangements. The main design features of the prototype SFR are double vessels (a reactor vessel and a guard vessel), a skirt-type core support structure excluding a weld joint with the reactor vessel, integral internal structures, a single rotating plug with a pantograph type IVTM (in-vessel fuel transfer machine), encompassed IHXs and primary pumps by a baffle plate, and a centre pivot structure preventing the dislocation of the core during normal operation.

Within the upper internal structures, thermocouples are installed to cover all fuel assemblies. The advanced concepts of the in-service inspection, repair and replacement, installations, and fuel transfer mechanism are considered in the early conceptual design stage.

3. RELATED R&D ACTIVITIES

3.1. Large scale sodium thermohydraulic test

A large scale sodium thermohydraulic test programme called STELLA (sodium test loop for safety simulation and assessment) is being carried out by KAERI. As the first step, the construction of the sodium component test loop called STELLA-1 has been completed, which is to be used for the thermohydraulic performance testing of major components such as heat exchangers, a mechanical sodium pump, and their design code verification and validation.

As the second step, an integral effect test loop called STELLA-2 will be constructed to demonstrate the plant safety and support the design approval for the prototype SFR. STELLA-2 is scheduled to be installed by 2016. The main experiments including the start-up tests will commence in 2017. Figure 6 shows the overall schedule of the STELLA project.

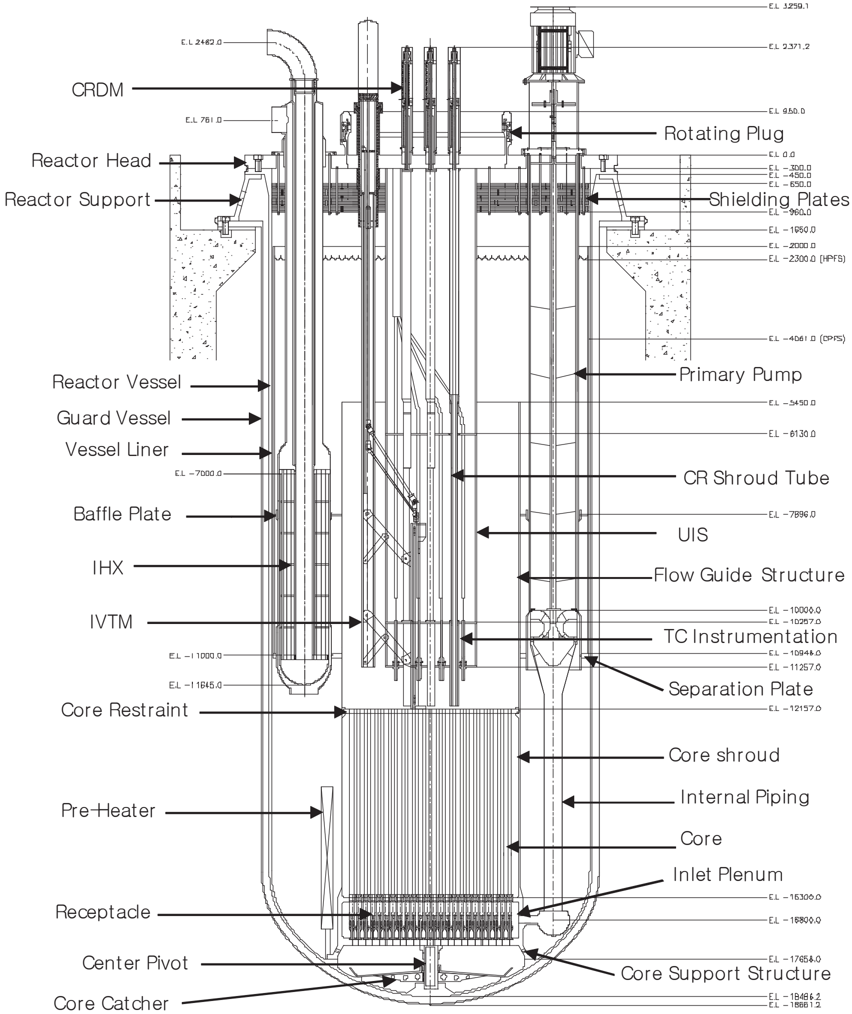


FIG. 5. PHTS arrangement of the prototype SFR.

STELLA-1 consists of a main test loop, a sodium purification system, and gas supply and related auxiliary systems. The main components of this facility are a sodium-to-sodium heat exchanger, a sodium-to-air heat exchanger, a mechanical sodium pump, loop heaters, a cold trap, a plugging meter, electromagnetic pumps, flowmeters, and a sodium storage tank. The component tests in STELLA-1 will start in 2013. The general specifications of the STELLA-1 facility are listed in Table 4, and a schematic diagram is shown in Fig. 7.

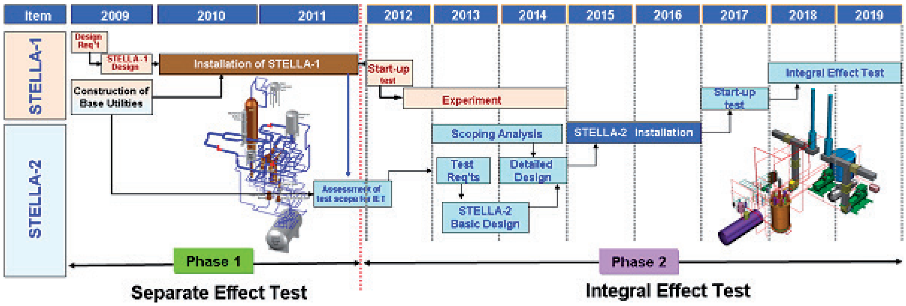


FIG. 6. Overall schedule of the STELLA project.

TABLE 4. STELLA-1 SPECIFICATIONS

Working fluid	Liquid sodium
Total electric power	2.5 MW
Total sodium inventory	~ 18 t
Heat capacity of HXs	1.0 MW
Design temperature	600°C
Design pressure	10 bar
Max. flow rate for HX test	10 kg/s
Max. flow rate for pump test	125 kg/s
Overall size (W×D×H)	15 m × 8 m × 22 m

3.2. Under-sodium viewing technique

The under-sodium viewing technique using an ultrasonic waveguide sensor has been developed for the under-sodium inspection of the reactor core and in-vessel structures in an opaque liquid metal — sodium. A novel under-sodium waveguide sensor module where beryllium (Be) and nickel (Ni) are coated on the SS304 waveguide plate has been suggested for the effective generation of a leaky wave in liquid sodium and the enhancement of a sodium wetting performance.

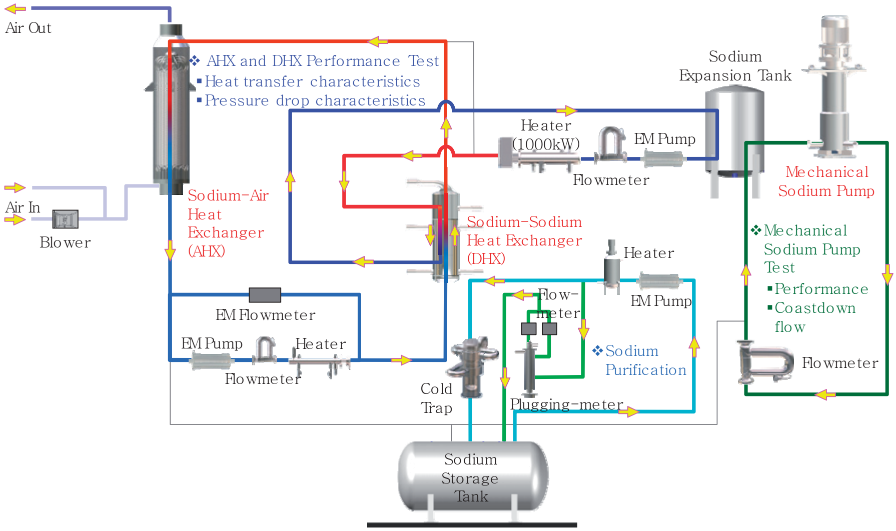


FIG. 7. Schematic diagram of STELLA-1.

The sensitivity of the under-sodium waveguide sensor module is evaluated by a measurement of the received ultrasonic signal from a flat reflector in sodium. Figure 8(a) shows the under-sodium C-scan test and Fig. 8(b) shows a typical ultrasonic pulse-echo signal, which has an initial pulse, a reflection signal from the end section of an under-sodium waveguide sensor, and a reflection signal from the test target in sodium (250°C).

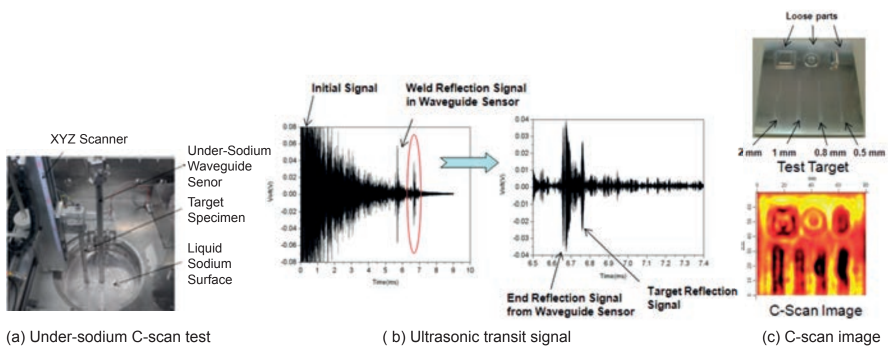


FIG. 8. Basic performance test of 10 m long under-sodium ultrasonic waveguide sensor in sodium.

The signal-to-noise (S/N) ratio of the reflection echo signal from the test target in sodium was measured at 10 dB. The visualization performance tests of the 10 m long under-sodium waveguide sensor module have been carried out by a C-scan test in sodium. The test target is a SS304 block with protruding defects and loose parts (block, washer and pin). As shown in Fig. 8(c), the loose part reflectors and protruding defects were clearly identified and resolved in the visualization image.

3.3. Metal fuel development

U–TRU–Zr metal fuel should be fabricated in a radiation shielded hot cell according to the high radioactivity of minor actinides, such as Cm and Am. Therefore, fuel fabrication technology with high reliability, simplicity and easy maintenance is being developed. To control the transport of volatile elements during the melting of a fuel alloy with minor actinides, a low pressure gravity induction casting system in which the melt in a crucible is cast into the mould under the crucible through a distributer and under gravity was designed and installed, as shown in Fig. 9 [4].

Other innovative fuel fabrication methods such as continuous casting and powder metallurgy are also under investigation. Optimization of the fuel casting process has been investigated to obtain the higher productivity of the sound fuel slugs. Using the advanced fuel casting system, U-10wt%Zr and U-10wt%Zr-5wt%RE (RE: Nd 53 wt%, Ce 25 wt%, Pr 16 wt%, La 16 wt%) fuel slugs were fabricated, as shown in Fig. 10. Gamma radiography was performed to detect internal defects such as cracks and pores inside the metal fuel slugs. The microstructures and thermomechanical property tests such as heat capacity, thermal expansion and high temperature tensile tests were measured. The capability of an advanced fuel casting system to control the volatile elements

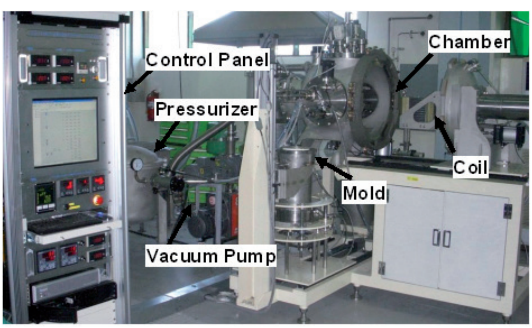
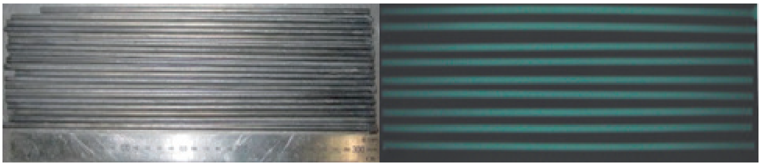


FIG. 9. Low pressure gravity casting system.



(a) (b)

FIG. 10. U-10wt%Zr-5wt% RE fuel slugs (a) and gamma radiography (b) fabricated by gravity casting method.

FIG. 11. HT9 cladding tubes.

during casting is being investigated using volatile elements such as manganese. Fuel slugs (U-10wt%Zr-5wt%RE-5wt%Mn) were fabricated without a significant loss of volatile manganese.

For the fuel rod cladding, HT9 cladding tubes were manufactured. A one tonne ingot was melted by vacuum induction melting process. The ingot was hot forged at 1200°C and a hollow billet was machined. The diameter of the hollow billet was 180 mm. A hot extrusion was introduced, and the diameter of the hot extruded tube was 54 mm. A hot extruded tube was pilgered. After pilgering, the drawing was carried out several times and an intermediate heat treatment was also performed after each drawing process. The final heat treatments were carried out at 1050°C and 750°C. The outer diameter, thickness and length of the cladding tubes were 7 mm, 0.6 mm and 3000 mm, respectively, as shown in Fig. 11. The mechanical tests such as the tensile, burst and creep tests have been performed with the HT9 cladding tubes. The tube fabrication processes such as the reduction ratio in the drawing and heat treatment conditions are also being investigated to improve the characteristics of the cladding tubes.

To study the interaction between the fuel slug and cladding such as the eutectic melting, diffusion couple tests were carried out by inserting barrier materials such as Zr, Nb, Ti, Mo, Ta, V and Cr between the fuel slug and cladding. Among these barriers, V and Cr exhibited the most promising performance. After scoping various coating methods, Cr electroplating was selected as one of the probable candidates because it is cost effective and easily applicable to a smaller tube geometry. However, it is revealed that when plating under a conventional condition, numerous cracks, which act as the diffusion path for the fuel component during the diffusion couple test, are generated during the plating. Further research will focus on reducing such cracks to enhance the Cr barrier performance. To demonstrate the barrier tube technology, 20 μm of Cr is uniformly plated at the inner surface of the 9Cr-2W FMS tube having a 4.6 mm inner diameter and 170 mm length.

Then, twelve fuel rodlets of U-Zr-(Ce) metal fuel with different fuel compositions in a single capsule were irradiated in HANARO up to 2.7at.% burnup. Figure 12 shows an irradiation capsule schematic diagram with coolant channel cross-sections. A post-irradiation examination of the irradiated fuels is currently being carried out in a hot cell. Representative examination parameters include the fuel burnup, fuel microstructure, swelling, fission gas release, constituent element redistribution, and behaviour of the Cr barrier.

3.4. Code development

3.4.1. Reactor physics experiment

To validate the neutronic characteristics of an SFR, KAERI has been collaborating with the Institute for Physics and Power Engineering (IPPE) in the Russian Federation. Three critical assemblies, BFS-73-1, -75-1, and -76-1A, were already constructed in BFS-1 or BFS-2 facilities. The first two critical

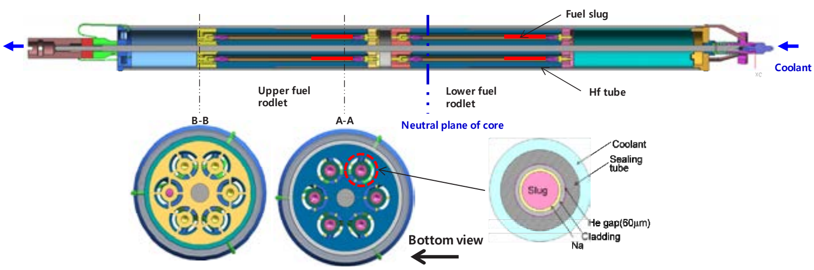


FIG. 12. Irradiation capsule for HANARO.

assemblies represented the early phase of the KALIMER-150 core design in the late 1990s. BFS-76-1A stands for the TRU burner concept, which is characterized by a core without a blanket, a low conversion ratio core, a high burnup reactivity swing, and a consequent deep insertion of a primary control rod at BOEC. As the representative of the current prototype SFR, the BFS-109-2A experiment is being conducted from 2012, which is constructed with highly enriched uranium (90wt%), depleted uranium, and zirconium metal to simulate U-10Zr metal fuels used in the prototype SFR.

3.4.2. Thermohydraulic validation tests of SFR computer codes

A number of in-house computer codes for the thermohydraulic design of an SFR are being developed at KAERI. Validation tests on the design codes are to be conducted in water and sodium environments. As part of thermohydraulic validation tests of the SFR design codes, three test activities are planned: (1) the validation tests of the reactor core thermohydraulic characteristics, (2) performance tests of the finned tube sodium-to-air heat exchanger (FHX), and (3) validation and verification (V&V) of the steam generator design code.

For the validation tests of the reactor core thermohydraulic characteristics, a test database on the core sub-channel flow characteristics in water and sodium environments is to be established through tests. The design concept of the facility for the sub-channel flow characteristics test in water has been developed. The concept of a finned tube heat exchanger in the SELFA (sodium thermohydraulic experiment loop for finned tube sodium-to-air heat exchanger) test loop has also been developed. The performance test for a scaled finned tube sodium heat exchanger will be carried out in the SELFA test loop. The test results will be used for the V&V of computer codes for FHX thermal sizing. The design of the experimental facility for the V&V of the computer codes analysing the heat transfer and pressure loss characteristics of the steam generator and the integrity evaluation is to be conducted by 2016.

3.4.3. Validation of SFR system analysis code

As a reliable tool for a system analysis of an SFR, KAERI has developed the MARS-LMR code by adding the SFR specific thermohydraulic models and neutronics models to the original version of MARS [5]. Recently, the applicability of MARS-LMR to a pool type SFR has been investigated, focusing on the influence of pool modelling. One of the important studies was the analysis of Phenix end-of-life natural circulation and asymmetry tests with different pool modellings. Through the analysis, it was found that the MARS-LMR code calculation with a one-dimensional pool modelling predicts the general trends of the tests successfully. It was also found that more accurate and realistic behaviours are obtained if the pool regions are modelled multi-dimensionally.

4. CONCLUSION

As one of the most promising technical options to resolve the spent fuel management problem, SFR technologies are being developed in the Republic of Korea. According to the national long term plan for SFR development, a conceptual design of the prototype SFR and related technologies for design support have been under development from 2012 with the goals of design approval by 2020 and construction by 2028.

From a trade-off study, the reactor capacity was determined to be 150 MW(e) with the use of U-Zr and U-TRU-Zr metal fuels for the early and later stages of reactor operation, respectively. The core outlet temperature is raised to 545°C to increase the plant thermal efficiency. The IHTS has two loops, and there are two IHXs connected to one steam generator and one IHTS pump in each loop. The decay heat removal system, one of the safety design features, is composed of two passive decay heat removal systems and two active decay heat removal systems. The mechanical structure design was carried out to obtain a simple reactor enclosure system, while applying advanced design technologies and materials. A safety analysis for the conceptual design will be performed in 2013.

ACKNOWLEDGEMENTS

This work was supported by Nuclear Research & Development Program of the National Research Foundation Grant funded by the Ministry of Education, Science and Technology in the Republic of Korea.

REFERENCES

[1] HAHN, D., et al., KALIMER Conceptual Design Report, Rep. KAERI/TR-2204/2002, Korea Atomic Energy Research Institute, Daejeon (2002).

[2] HAHN, D., et al., KALIMER-600 Conceptual Design Report, Rep. KAERI/TR-3381/2007, Korea Atomic Energy Research Institute, Daejeon (2007).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Fast Reactor Database 2006 Update, IAEA-TECDOC-1531, IAEA, Vienna (2006).

[4] LEE, C.T., et al., "Casting Technology Development for SFR Metallic Fuel," paper presented at Int. Conf. GLOBAL: The Nuclear Fuel Scycle: Sustainable Options and Industrial Perspectives, Paris, France, 2009.

[5] HA, K.S., et al., "Development of MARS-LMR and Steady-state Calculation for KALIMER-600," KAERI/TR-3418/2007, Korea Atomic Energy Research Institute, Daejeon (2007).

ENHANCEMENT OF THE JSFR SAFETY DESIGN AND CRITERIA FOR THE GEN-IV REACTOR

K. AOTO, Y. CHIKAZAWA, T. OHKUBO,
Japan Atomic Energy Agency,
Ibaraki

K. OKADA, T. ITO
Mitsubisi FBR Systems Inc,
Tokyo

Japan

Abstract

An overview is provided of Japan's sodium cooled fast reactor (JSFR) development status and reflection on lessons learned from TEPCO's Fukushima Dai-ichi nuclear power plant (1F). The JSFR was recognized as a promising next generation nuclear reactor. Even though the JSFR safety design already took into account measures against severe accident situations and passive safety features, such as passive shutdown systems and natural convection decay heat removal systems in the 2010 design version, it has become aware of the importance of design measures against severe accidents and extreme external events by the 1F accident. As recent activities, external hazard evaluations and design improvements reflecting lessons learned from 1F accident have been conducted. This paper also discusses importance of development of global safety design criteria and international R&D cooperation on safety design measures.

1. INTRODUCTION

Japan's sodium cooled fast reactor (JSFR) is a concept which has the potential to achieve sustainable energy production, radioactive waste reduction and safety equal to future LWRs and an economically competitive energy source when measured against other future energy sources. In 1999, the JAEA and utilities launched the Feasibility Study on Commercialized Fast Reactor Cycle Systems (FS) with domestic partners comprising vendors and universities. The FS targets aim at improved safety and economic competitiveness looking at other energy resources in the future including the future LWR [1, 2]. These targets are consistent with the goals of Generation IV International Forum (GIF). For safety, the targets are as follows:

- Fundamental safety principles will be observed. Safety standards and guidelines for former SFRs will be kept while specific features of new reactors will be considered.
- Prevention and mitigation against severe accident initiators will be considered so as to avoid exercise of the off-site emergency plan.
- Total CDF will be less than 10^{-6}/reactor-years in the case of multiple units on a site, and the total frequency of loss of containment function in core damage conditions will be less than 10^{-7}/reactor-years.
- Adequate maintenance/repair rule will be developed and the design concept will be well fitted with this rule. Inspection devices will be suitably developed.

The FS scope includes various advanced FRs: SFR, gas cooled FR (GFR), heavy metal cooled FRs (lead cooled FR (LFR) and lead–bismuth cooled FR (LBFR)), water cooled FRs with various fuels such as oxide, nitride and metal. Evaluation showed that SFR, GFR and LFR/LBFR could achieve the sustainability target but LFR/LBFR and GFR would require an experimental FR and have to overcome several issues with high difficulty. From the viewpoint of SFR configuration, the loop type concept was selected since it succeeded in dramatically reducing the construction cost from conventional loop concepts and it could have the higher potential on the construction cost reduction compared to the advanced pool concepts such as the EFR [3]. As a result, the advanced loop type SFR, named JSFR, was selected as the Japanese FR concept.

The USA–France–Japan joint paper presents a common view of FR specific missions in the development of nuclear energy and a cross-analysis of the merits and demerits of several FR concepts studied worldwide and especially in the GIF framework. This trilateral study confirmed that the fundamental mission of the FR was to achieve significant uranium utilization and waste management goals. Thus, fast spectrum reactor concepts are vital for nuclear fuel cycle sustainability goals. The three countries agreed that the SFR, GFR and LFR were capable of achieving these goals. However, the SFR is the most matured technology from the viewpoint of industrial deployment while both GFR and LFR still require long term development before an experimental reactor project could begin [4].

The three countries agree that the SFR has significant proven base technologies and a clear understanding of the remaining challenges to be addressed before industrial deployment. The most important advantages of the SFR are accumulation of fuel and safety experiences based on past and existing experimental and prototype SFRs. From the safety point of view, many large demonstration experiments have been conducted and evaluation tools validated by these experiments have been developed. Phenomena in the case of CDA were well analysed by these evaluation tools. After the Fukushima accident, understanding and evaluation of CDA has become more and more important. SFR measures against CDA already reach the demonstration phase. In several reactors, demonstrations of natural convection decay heat removal operation have been successfully conducted. Safety performances of future SFR will take benefit of this important feedback on the one hand and have to comply with the stringent objectives of the GEN IV on the other hand [4].

In this paper, an overview of JSFR development status, including system identification, key technology evaluations and safety design features in the 2010 version, has been summarized. In recent activities, external hazard evaluations and design improvements reflecting lessons learned from 1F accident have also been summarized. This paper also focuses on the importance of development of global safety design criteria and international R&D cooperation on safety design measures.

2. JSFR SAFETY DESIGN IN 2010

In the FaCT Phase I, ten major JSFR key technologies, as shown in Table 1, have been evaluated [5]. All of these technologies have been evaluated as being suitable for installation in both demonstration and commercial JSFR. As for the ODS steel, a further evaluation on manufacturing with stable quality will be re-evaluated.

For safety design [6, 7], the JSFR adopts the defence-in-depth (DiD) principle to the same extent as in LWRs. The deterministic approach was also adopted considering design-basis accidents (DBAs) to specify safety functions such as a reactor shutdown system (RSS) and a decay heat removal system (DHRS) for prevention of core damage. The JSFR has several design measures against severe accidents explicitly taking into account accidents such as design extension conditions (DECs). In addition to the DiD principle, the JSFR also adopts a risk informed approach that plays a role in considerations on the proportion or balance of different levels of DiD.

Securing reactor shutdown, two independent reactor shutdown systems (RSS), i.e. primary and backup RSS, are installed. Each RSS is initiated by independent/diversified signals from the reactor protection system. The fourth level of DiD considers design measures against DECs. In this level, including both prevention and mitigation of severe accidents, the RSS provides passive shutdown capability installing an SASS (self-actuated shutdown system). Performances of SASS had already been confirmed through the transient experiments in the sodium loop and reliability testing has been achieved by installing SASS mock-up at Joyo [8].

TABLE 1. JSFR KEY TECHNOLOGIES

Plant constituent parts	Key technologies
Core and fuel	High burnup fuel with ODS cladding material Safety enhancement technologies; SASS, re-criticality free core
Reactor system	Compact reactor system
Cooling system	Two-loop cooling system of large diameter piping made of Mod. 9Cr-1Mo steel Integrated pump-IHX component Highly reliable SG with double-walled straight tube
DHRS	Capability of decay heat removal by natural circulation
BOP	Simplified fuel handling system
Reactor building	CV made of steel plate reinforced concrete (SCCV) Advanced seismic isolation system for SFR

The re-criticality free core concept has great importance in ensuring the in-vessel retention scenario against whole core disruptive accidents. The initiating phase energetics due to exceeding the prompt criticality has to be prevented to restrict the sodium void worth and the core height. The possibility of molten fuel compaction has to be eliminated by enhancing the fuel discharge from the core. The effectiveness of fuel assembly with inner duct structure (FAIDUS) has been confirmed by both in-pile and out-of-pile experiments [9].

The JSFR decay heat removal system (DHRS) consists of a combination of one loop of direct reactor auxiliary cooling system (DRACS) and two loops of primary reactor auxiliary cooling system (PRACS) adopting a full natural convection system, as shown in Fig. 1. The heat exchanger of DRACS is dipped in the upper plenum within the reactor vessel. The heat exchanger of each PRACS is located in the primary-side upper plenum of an intermediate heat exchanger. The DHRS can be operated by a fully passive feature with natural convection, which requires no active components such as pumps [10]. The JSFR DHRS

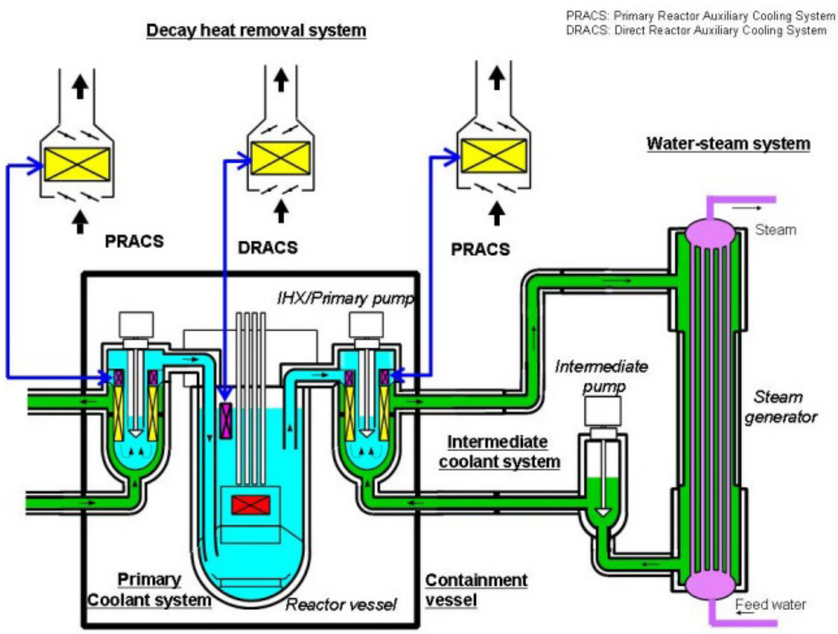


FIG. 1. JSFR DHRS configuration in 2010 [7].

was confirmed by a 1/10 scale water test and evaluation tools were verified and validated by the experimental data [11, 12]. From the view point of probabilistic safety assessment, thanks to the passive features, frequency of protected loss of heat sink (PLOHS) was evaluated to be 2 × 10^{-8}/reactor-years [13] and less than 10^{-8} taking into account accident management (AM) [5].

Since the JSFR adopts a full natural convection DHRS, it is free from heavy electric load and quick activation of emergent electric supply. This means that the JSFR is capable of employing a self-air-cooling gas turbine (GT) independent of the component cooling water system (CCWS) [14]. In fact, the JSFR CCWS is non-safety grade thanks for the natural convection DHRS and self-air-cooling GT. This configuration reinforces defence against external hazards. In the case of external hazards such as tsunami, the CCWS could be damaged as in the Fukushima accident, since the heat sink of the CCWS depends on seawater.

For seismic design, the JSFR adopts an advanced seismic isolation system, which mitigates the horizontal seismic force by thicker laminated rubber bearings with longer period and improves damping performance by adopting oil dampers. To confirm feasibility of the thicker laminated rubber bearings, a basic characteristic test with a 1/8 reduced scale model was carried out. As a result, the possibility of the application of the thicker laminated rubber bearings was confirmed [15].

3. LESSONS LEARED FROM THE FUKUSHIMA ACCIDENT

The 1F accident, caused by the Great East Japan Earthquake of 11 March 2011, recalls the need for ensuring that there are sufficient design measures against severe accidents and extreme external events.

Study of lessons learned from the 1F accident is being conducted. The first point is the enhancement to systems that may be needed in order to decrease the likelihood of a severe accident due to extreme external hazards. Namely, robustness should be enhanced in power supplies (DC and AC, if needed to power active safety system), cooling functions (core, CV and spent fuel pools), and heat transportation system including final heat sink. The second point is the enhancement of response measures against severe accidents. The means should be provided to prevent severe mechanical loads on CV and the instrumentation should be prepared to identify the status of reactor core and CV. The third point is the reinforcement of safety infrastructure by ensuring independency and diversity of safety systems.

Provision for external events needs to be sufficiently robust in coordination with anticipated conditions at the reactor site. For example, the design must consider ensuring power supply during long term loss of all AC power. Enhancing passive functions will reduce the dependency on power supply and it is also effective as a measure against power loss. External events such as earthquakes, tsunami and floods may become initiators of SA, therefore required protection measures should be provided as well as ensuring margin. Stress tests are one possible method to evaluate the safety margins of the NPPs beyond the design basis.

From the viewpoint of toughness against external events, the JSFR had already improved safety features as a next generation reactor at a pre-conceptual design version in 2010. JSFR toughness against earthquake and tsunami has been evaluated based on the 2010 design version [16].

To prove JSFR safety tolerance against external hazards, external hazard conditions have been comprehensively investigated [17] and safety evaluations on the demonstration JSFR version 2010 have been conducted. Comparison on seismic conditions has shown that the seismic conditions used in the 2010 design study are more stringent than the 1F conditions. In the reference study [17], seismic analyses were conducted with a 1F-enveloping acceleration spectrum regarding the 1F ground conditions and the analyses showed that the JSFR had sufficient design margin for shutdown capability and integrity of major components against the severe seismic conditions enveloping the 1F conditions.

As the 1F accident, the seawater pumps for the CCWS could be totally damaged, since they were located the sea level. Then, the CCWS could fail due to tsunami, since it depends seawater as a final heat sink. In the JSFR design, safety components, including DHRS and emergency power supply, are independent from the CCWS thanks to full natural convection DHRS and air cooling GT. Even in a hypothetical SBO, decay heat could be removed by a natural convection DHRS. A transient of hypothetical SBO has been analysed to evaluate the time margin to AC freezing as shown in Fig. 2. On the reactor trip, one DRACS and two PRACSs are activated and DRACS/PRACS dumpers are controlled by power supply from the emergency batteries. After 2 hours, the DRACS and PRACS damper control is lost due to battery capacity and the damper openings are assumed to be maintained as is. In the analytical case, the decay heat and AC air inlet temperature are assumed to be low and the AC dumpers are, conservatively, assumed to be fully opened. As a result, AC freezing in DRACS and PRACS happens in 7 hours and 12 hours after the reactor trip, respectively. After the loss of DHRS, the RV temperature could reach 650°C in 25 hours. The result shows that the JSFR has enough time margin for accident management of manual AC dumper control. However, the result also shows that an additional emergency power supply and DHRS with diversity could enhance further prevention of PLOHS potential regarding failure of manual AC dumper control [16].

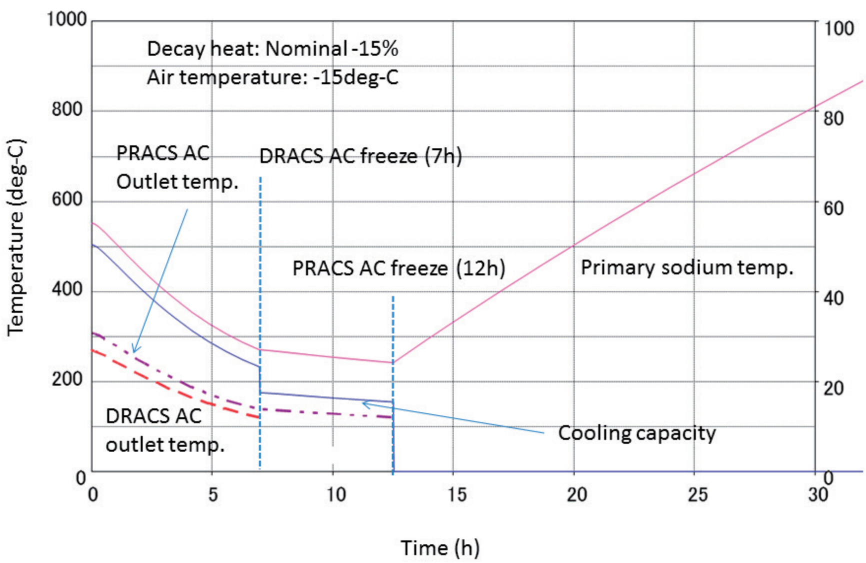


FIG. 2. Transient of hypothetical SBO [16].

For the fuel handling system, seismic analysis and toughness against hypothetical SBO has also been evaluated. Seismic analysis showed that fuel handling components and fuel storages had enough seismic margins. For fuel storage cooling, the JSFR has an ex-vessel storage tank (EVST) and spent fuel water pool. The ex-vessel storage has a redundant sodium cooling system and the final heat sink is air. The analyses showed the capability of natural convection cooling of EVST and enough time margin (more than one week) to loss of spent fuel water pool [18].

In conclusion, the JSFR already has external hazard toughness, even in the 2010 design version, mainly thanks to the passive safety features and seismic isolation system. However, as shown in hypothetical analysis, additional design improvements still have the potential to reduce CDA frequency, and the requirements of these additional measures should be tested a new safety approach regarding lessons learned from the 1F accident.

4. SAFETY APPROACH

The GIF high level safety standard is written in Basis for Safety Approach for Design & Assessment of Generation-IV Nuclear Systems [19]. However, there is a large gap between the high level safety standards and actual safety designs. Safety design criteria (SDC) are being prepared as middle level global safety standards [20]. The SDC provide specific safety criteria in general and for each component, such as fuel, core, cooling system and containment in the SFR system regarding the GIF safety goals, SFR safety features, innovative technologies and lessons learned from the 1F accident.

Fast reactors, including the SFR, have the characteristic that a core is not in the most reactive configuration, and thus they have the possibility to result in positive reactivity changes when assuming sodium boiling, clad discharge and fuel concentration resulting from a core disruptive accident (CDA). Handling of a CDA is focused traditionally in the SFR, however, approaches and identification of initiating events depend on the authorities in each country. The SDC proposes that all possible initiating events should be identified systematically and comprehensively, combining deterministic and probabilistic approaches. Exclusion of initiating events should accompany technically supported justification. As lessons learned from the 1F accident have shown, external hazards, including human induced events, should also be included when identifying initiating events.

Mitigation of the consequences of some accident situations should be excluded by design where feasible. Several candidates for situations to be practically eliminated have been identified, which are selected based on the design characteristics of the SFR [21]. Among these candidate situations, anticipate transient without scram (ATWS) and loss of heat removal system (LOHRS) are especially important, since they strongly affect core/plant designs.

In ATWS type events, in-balance of power and cooling might cause core damage within a shorter time period. Passive shutdown mechanisms such as SASS can prevent core damage even under such conditions. In addition, mitigation of core damage is considered in the design because of the shorter time period to core damage and the potential mechanical energy release, which might appear in core damage situations. Then, the practically eliminated situation for ATWS should be "severe mechanical energy release due to coherent sodium boiling or molten fuel compaction, failure of decay heat removal from degraded core in ATWS type events".

LOHRS includes loss of coolant flow (flow path disruption, etc.), loss of reactor revel (LORL) and protected loss of heat sink (PLOHS). Since the decay heat is at a few per cent of the nominal power, the temperature of the reactor coolant system would be slow, with sufficient time margin to make recovery action for failed DHRSs and/or implementation of backup cooling measures. If no heat sink is available, core and coolant boundary failure might occur due to the high temperature causing significant thermal loads not coped with by the containment. Then, the practically eliminated situation for LOHRS should be "significant core damage in LOHRS type events".

Reference [20] identifies several candidates of practically eliminated situations, i.e. those situations that have to be screened and tested under international consensus and criteria. The SDC would help and be the basis for the international consensus and criteria.

5. SAFETY DESIGN IMPROVEMENTS

For ATWS, design measures confirmed in FaCT phase I are effective and sufficient against the new SDC. For CDA prevention and mitigation, the JSFR adopts SASS and FAIDUS, respectively, to practically eliminate severe mechanical energy release due to ATWS. Regarding these design measures, the JSFR aims for in-vessel retention. The ATWS prevention and mitigation technologies have been already matured to allow large scale demonstration experiments to be conducted and these experiments require international partnership. The ATWS scenario and prevention/mitigation measures should be shared and standardized to cross-check effectiveness of these measures and evaluation tools.

For LOHRS, the SFR has superior characteristics thanks to sodium coolant features such as low pressure and high natural convection capability. Utilizing these superior characteristics of sodium, the JSFR already equipped reliable DHRS with natural convection which does not depend on emergency AC power and the PLOHS frequency is much lower than the target of 10^{-6}/reactor [5]. The lessons learned from the 1F accident pointed out the importance of the cliff edge effect. Practical elimination of LOHRS is important from the viewpoint of the in-vessel retention, since primary coolant boundary failure due to high sodium temperature precedes significant core damage in the case of LOHRS. Even the original JSFR DHRS configuration has sufficient reliability against all possible situations; installation of additional DHRS with independency and diversity could improve toughness against LOHRS. Figure 3 shows candidates of additional DHRS options under discussion. DRACS and PRACS are systems for DBA and circulated by fully natural convection. The DRACS will be improved to cover siphon break in the primary sodium circuits and low reactor sodium level. For additional DHRS, the auxiliary core cooling system (ACS) and SG gas cooling system have been identified as candidates to improve DHRS diversification and effectiveness in severe accident management. The study on diversified DHRS also covers the reactor vessel cooling system as one of the backup options, as shown in Fig. 4. These design options should be compared from the viewpoint of feasibility, diversification, reliability and cooling performance to meet requirement of the practical elimination of LOHRS.

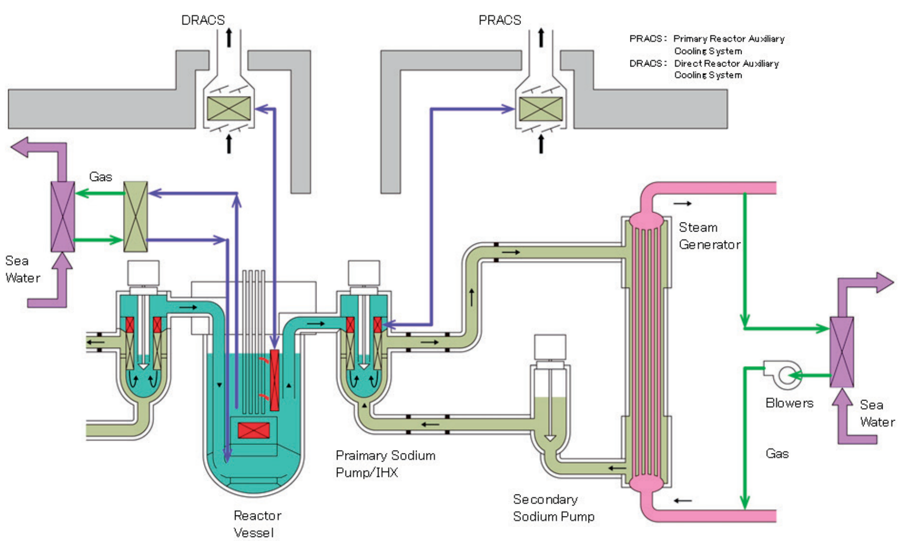


FIG. 3. JSFR DHRS options.

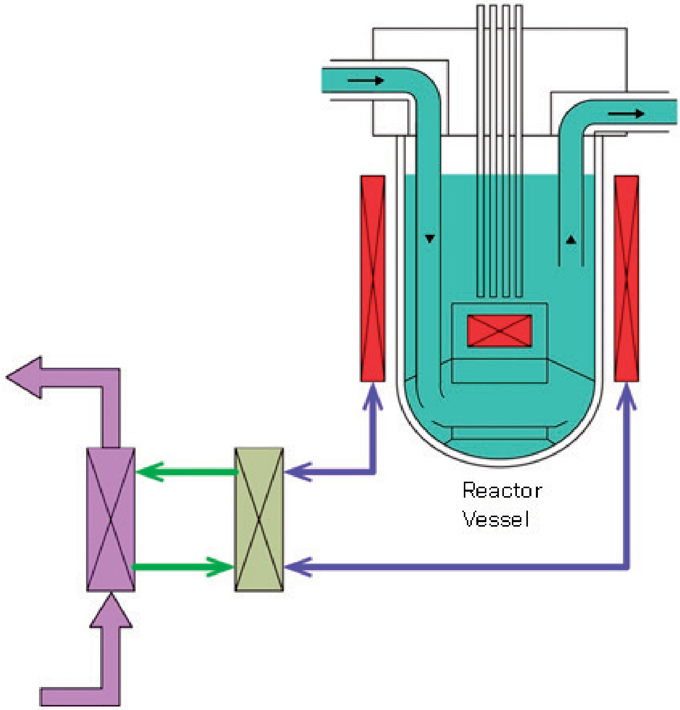


FIG. 4. DHRS backup option.

6. INTERNATIONAL COOPERATION ON LOHRS MEASURES

To accumulate experimental data on core cooling in severe conditions, the JAEA proposes international cooperation in providing sodium loop facilities. International cooperation on LOHRS is also important, since practical elimination of LOHRS is essential for the IVR in the SFR. Criteria for practical elimination of LOHRS and sufficient design measures should be discussed internationally and design measures and design tools should be developed and validated internationally regarding objectivity and cross-checking. The JAEA has constructed the AtheNa facility with a large sodium loop with a total sodium inventory of 240 t, as shown in Fig. 5, and also has existing middle scale sodium loops such as PLANDTL with sodium inventory 25 t. Using these sodium loops, the JAEA can provide experimental test beds for international cooperation on DHRS development. Figure 6 shows an example schematic of AtheNa test loops for a 1/3 scale reactor vessel cooling experiment. The reactor vessel cooling adopted by past French and US designs such as Superphenix and ALMR [22],

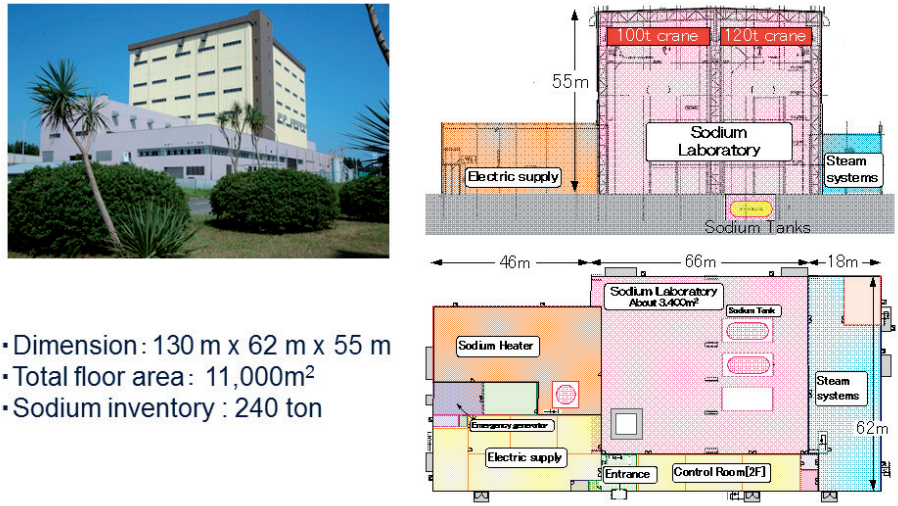


FIG. 5. Overview of AtheNa facility.

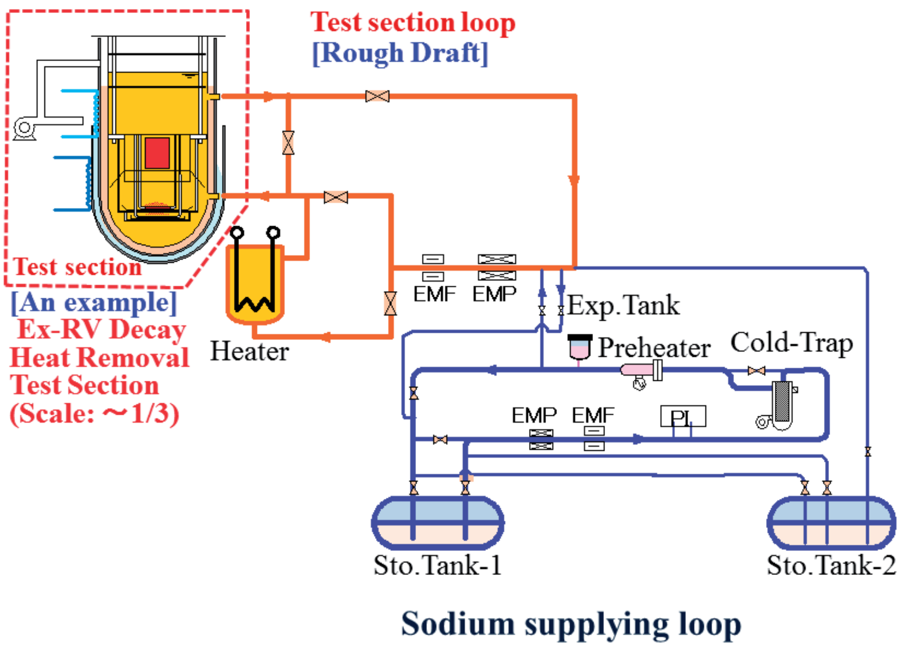


FIG. 6. Schematic of the AtheNa loop (example).

are candidates for international cooperation. The AtheNa facility also has the capability to conduct experiments on other types of DHRS such as SG gas cooling using a SG mockup which has already been fabricated. Through these experiments, advanced DHRS and design tools would be developed and provide the technological basis for next generation SFR safety.

7. CONCLUSIONS

An overview of JSFR development status has been given and reflection of lessons learned from the 1F accident summarized. The JSFR was recognized as a promising next generation nuclear reactor through FS. The joint feasibility study between the USA–France–Japan also confirmed that sodium coolant was the most mature technology to establish a sustainable nuclear energy system. Even though the JSFR safety design already takes into account measures against severe accident situations and passive safety features, such as a passive shutdown system and natural convection decay heat removal systems, as in the 2010 design version, the 1F accident signalled the importance of design measures against severe accidents and extreme external events. For external hazards, evaluations have been conducted and shown that the JSFR already had sufficient safety design features against severe earthquakes and the long term loss of all AC power. Furthermore, the JSFR's DHRS is being improved, adding additional DHRSs such as ACS and SG gas cooling to practically eliminate LOHRS. Additionally, international cooperation has become especially important after the 1F accident, since safety design criteria/measures should be tested with objectivity and under cross-checking. For further improvement on safety designs based on lessons learned from the 1F accident, SDC is under discussion in the GIF framework aiming at global standards for testing SFR safety designs. International R&D cooperation on DHRS in severe situations has been proposed. The JAEA AtheNa facility can serve as an experimental test bed for various DHRS performance tests using large sodium loops.

ACKNOWLEDGEMENTS

This paper includes results of "Technical development program on a commercialized FBR plant" entrusted to JAEA by the Ministry of Economy, Trade and Industry of Japan (METI).

REFERENCES

[1] AIZAWA, K., et. al., "R&D for Fast Reactor fuel cycle technologies in JNC", Proc. Int. Conf. GLOBAL, Paris, France, 2001 (2001) CD-ROM.

[2] NODA, H., INAGAKI, T., "Feasibility study on commercialized FR cycle systems in Japan — The results in the first phase and future plans of the study", Proc. Int. Conf. GLOBAL, Paris, France, 2001 (2001) CD-ROM.

[3] CHIKAZAWA, Y. KOTAKE, S., SAWADA, S., "Comparison of Pool/Loop Configurations in the JAEA feasibility study 1999-2006", Proc. Int. Conf. on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), Kyoto, Japan, 2009, IAEA, Vienna (2012).

[4] SAKAMOTO, Y.,GARNIER, J.C., ROUAULT, J., GRANDY, C., FANNING, T., HILL, R., CHIKAZAWA, Y., KOTAKE, S., Selection of sodium coolant for fast reactors in the US, France and Japan, J. Nucl. Eng. Des. **254** (2013) 194–217.

[5] CHIKAZAWA, Y., AOTO, K., HAYAFUNE, H., OHNO, Y., KOTAKE, S., ITO, K., TODA, M., "Conceptual design for a large-scale Japan sodium-cooled Fast Reactor (1) feasibility of key technologies", Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP), Nice, France (2011) 11278.

[6] KOTAKE, S., UTO, N., AOTO, K., KUBO, S., "Safety design features for JSFR — Passive safety and CDA mitigation", Proc. Annual Meeting on Nuclear Technology, Dresden, Germany, 2009.

[7] KUBO, S. et al., "Safety design requirements for safety systems and components of JSFR", Proc. Int. Conf. on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), Kyoto, Japan, 2009, IAEA, Vienna (2012).

[8] TAKAMATSU, M., et al., Demonstration of control rod holding stability of the self actuated shutdown system in Joyo for enhancement of Fast Reactor inherent safety, J. Nucl. Sci. Technol. **44** (2007) 511.

[9] SATO, I., et al., "Elimination of severe recriticality events in the Core Disruptive Accident of JSFR aiming at In-Vessel Retention of the core materials," Proc. Int. Conf. on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), Kyoto, Japan, 2009, IAEA, Vienna (2012).

[10] YAMANO, H., KUBO, S., KURISAKA, K., SHIMAKAWA, Y., SAGO, H., Technological feasibility of two-loop cooling system in JSFR, J. Nucl. Technol. **170** (2010) 159.

[11] OHYAMA, K., et. al., "Decay heat removal system by natural circulation for JSFR", Proc. Int. Conf. on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), Kyoto, Japan, 2009, IAEA, Vienna (2012).

[12] KAMIDE, H., et al., "Sodium experiments on decay heat removal systems of Japan Sodium Cooled Fast Reactor – Start-up transient of decay heat removal system", Proc. 7th Korea-Japan Symposium on Nuclear Thermal Hydraulics and Safety , Chuncheon, Korea (2010) N7P0057.

[13] KURISAKA, K., "Probabilistic safety assessment of Japanese Sodium Cooled Fast Reactor in conceptual design stage", paper presented at 15th Pacific Basin Nuclear Conference (PBNC-15), Sydney, Australia (2006).

[14] HISHIDA, M., KONOMURA, M., UCHITA, M., IITSUKA, T., KAMISHIMA, Y., "Construction of Sodium-cooled Medium-scale Modular Reactor in Consideration of In-servive Inspection and Repair", Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP), Seoul, Korea (2005) 5112.

[15] IKAMURA, S., ETOH, M., KAMISHIMA, Y., NEGISHI, K., SAKAMOTOM, Y.. KITAMURA, S., KOTAKE, S., "Seismic Isolation Design for JSFR", Proc. Int. Conf. on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), Kyoto, Japan, 2009, IAEA, Vienna (2012).

[16] HAYAFUNE, H., "Evaluation of severe external events on JSFR", paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[17] YAMAMNO, H., KUBO, S., TANI, A., NISHINO, H., SAKATI, T., "Safety design approach for external events in Japan Sodium-cooled Reactor", Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP), Chicago, IL (2012) 12158.

[18] KATOH, A., CHIKAZAWA, Y., UZAWA, M., "Evaluation and Improvement on External-hazard proof of JSFR Fuel Handling System", Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP), Chicago, IL (2012) 12390.

[19] GENERATION IV INTERNATIONAL FORUM RISK & SAFETY WORKING GROUP, Basis for safety approach for design & assessment of Generation-IV Nuclear Systems, Rep. GIF/RSWG/2007/002 (2008).

[20] NAKAI, R., OKANO, Y., KUBO, S., "Development of safety design criteria for the Generation-IV Sodium-cooled Fast Reactor", Proc. of 8th Japan-Korea Symposium on Nuclear Thermal Hydraulics and Safety (NTHAS8), Beppu, Japan (2012).

[21] KUBO, S., "Safety Design Approach for JSFR toward the realization of Gen IV Sodium Cooled Fast Reactor", paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[22] INTERNATIONAL ATOMIC ENERGY AGENCY, Fast Reactor Database — 2006 Update, IAEA-TECDOC-1531, IAEA, Vienna (2006).

DEVELOPMENT OF PHYSICAL CONCEPTIONS OF FAST REACTORS

YU.S. KHOMYAKOV, V.I. MATVEEV, A.V. MOISEEV
Institute for Physics and Power Engineering,
State Scientific Center of the Russian Federation,
Obninsk, Russian Federation

Abstract

A unique experience has been gained in the Russian Federation in developing the fast reactor concept starting with its appearance in 1950s and 1960s: from BR-1 first critical facility, BR-10 and BOR-60 experimental reactors and BN-350 semi-commercial reactor to the BN-600 commercial reactor in Beloyarsk NPP. Based on this experience the analysis of motivation of development of fast reactor physics concepts and modern Russian requirements for innovative reactor designs is presented in the paper. Key scientific and technical problems related to development of the new generation fast reactors are stated, and adopted approaches to the following main areas of studies are discussed: (1) increase of core self-protection level, (2) improvement of technical and economic characteristics, (3) assurance of nuclear power fuel supply and assimilation of closed nuclear fuel cycle, and (4) destruction of long lived radioactive waste and transmutation of minor actinides. Two basic trends in the development of advanced reactor designs in the Russian Federation (fast reactors with sodium and heavy liquid metal coolant) are outlined.

1. OUTLINE OF RUSSIAN EXPERIENCE IN MASTERING FAST REACTORS

The history of fast reactor development in the Russian Federation testifies to systematic and consequential mastering of this reactor technology. In each stage, there were key scientific and technological problems, their solutions being taken into account in determining the objectives of the next stage. Below are presented the most important stages of development and mastery of fast reactor technology in the Russian Federation and their basic results [1].

1.1. BR-1 critical facility

BR-1 was the first zero power fast reactor in the Russian Federation and in Europe using plutonium metal [2, 3]. The key result obtained in this reactor was experimental confirmation of the basic physical concept of fast reactors, namely, the possibility of fuel breeding with a unique breeding ratio (BR = 2.5 ± 0.2).

It should be noted that this high breeding ratio (which is much higher than those reached up to now ~1.0–1.5) strongly influenced the physical concept of the first generations of fast reactor. The designers strived after using the above resource to the maximum extent in order to produce new fuel, which was non-existent in nature before, i.e. Pu-239 in the amount sufficient for nuclear power development in a relatively short time.

1.2. BR-5/10 — the first sodium cooled reactor

Based on BR-1 experience, the BR-2 experimental reactor of 100 kW power was put into operation in 1956. Its core with plutonium metal fuel was cooled by mercury [4]. Operation of this reactor demonstrated the lack of prospects of mercury coolant, mainly because of its toxicity and low compatibility with structural materials.

In 1957, the BR-2 reactor was dismantled and the BR-5 reactor was constructed as its replacement. In this reactor sodium coolant was used for the first time. Startup and operation of the BR-5 reactor made it possible to gain experience in the large scale testing of the fuel elements, including failed fuel elements and the real experience of sodium systems operation. The most important physical measurements of transients dynamics, as well as temperature and power reactivity effects, were carried out for the first time.

On the basis of the results of BR-10 reactor operation, it was recommended that the use of plutonium metal fuel be discontinued and a switch made to ceramic fuels. In the course of tests, the following maximum fuel burnup values were achieved [1]: plutonium oxide fuel ~6.7% h.a.; uranium mononitride fuel ~9% h.a.; and uranium monocarbide fuel ~6% h.a.

1.3. BN-350 semi-commercial reactor

The BN-350 reactor of 1000 MW(th) and 350 MW(e) was the world’s first prototype fast reactor. The first criticality of this reactor was gained in November 1972, and it was connected to the grid on 16 July 1973. The BN-350 was a dual-purpose facility, it was designed for both energy production and seawater desalination (capacity of desalination plant was 120 th. m^3 per day). Loop design was used for this reactor plant; heat was removed from the core by six parallel sodium loops.

The BN-350 experience produced many unexpected technological results. Problems related to evaporators of the steam generators in the early stage of reactor operation jeopardizing the feasibility of fast reactors were successfully solved.

Such problems as the considerable radiation swelling of structural steels and absence of gas plenums in the fuel elements manifested themselves to the full extent. Designs of both fuel subassemblies and fuel elements were significantly upgraded in the course of operation. In the BN-350, reactor tests were carried out on methods of power rate profiling (using 2 and 3 fuel enrichment zones in the core) and control rods with various materials (B10, Eu_2O_3 and neutron traps with ZrH_x).

1.4. BN-600 commercial reactor

Successful operation of the BN-600 reactor since its startup in 1980 in the Beloyarsk NPP [5] up to now is a rather major success for the Russian Federation in the area of fast reactors. This is currently the only commercial fast reactor worldwide which is operating consistently and generating electricity as part of a commercial NPP [6]. The BN-600 is the first pool design reactor in the Russian Federation.

During BN-600 operation, work on the measurement and optimization of reactor parameters was continually carried out. After revealing the high extent of fuel element failures in the first core loading, some modifications were made to the core in 1986–1988, including: core height was increased from 75 to 103 cm, linear power of the fuel elements was decreased from 54 to 48 kW/m, three-zone power profiling was introduced instead of the two-zone scheme and shuffling of the fuel subassemblies from the periphery of the core towards its centre during reactor refuelling procedures was abandoned.

In 1992–1994, maximum fuel burnup of 10% h.a. was achieved with certainty. In the 2004–2006 period, owing to optimization of the reactivity balance and confirmation of reliability of the fuel elements, a threefold refuelling mode was replaced by the fourfold mode, and maximum fuel burnup was increased up to 11.2% h.a. [7].

2. BASIC TRENDS OF DEVELOPMENT OF PHYSICAL CONCEPTIONS OF FAST REACTORS

Owing to the successful operating experience of the pool type design of the BN-600 reactor, it has become a classic and is used as the basis for the new fast reactors, such as the BN-800 and BN-1200. In the Russian Federation, construction of the new commercial BN-800 reactor is proceeding to completion. Putting the BN-800 into operation is associated with the beginning of formation and development of the closed fuel cycle system in the Russian Federation, since this will be the first fast reactor using mixed uranium–plutonium fuel.

Nevertheless, neither BN-600 nor BN-800 is considered in the Russian Federation as the reactor prototype for the new generation NPP.

What is stopping further use of existing fast reactor designs? There are several unresolved problems:

— Safety: increase of self-protection level of the core;
— Economy: improvement of technical and economical characteristics;
— Resources: assurance of fuel supply of nuclear power and assimilation of closed nuclear fuel cycle;
— Radioactive wastes: transmutation of minor actinides.

The key problems (“choice forks”) to be solved for making conceptual technological decisions on the new generation reactors are as follows:

— Evaluation of feasibility of inherently safe fast reactors;
— Choice of coolant: sodium, heavy liquid metal, gas or steam;
— Choice of fuel type: MOX, carbide, nitride or metal;
— Fuel breeding level: from BR ~1 to BR ~1.5;
— Fuel breeding level in the core, optimality of the core with equilibrium fuel and BR core ~1;
— Expediency of use of fertile blankets;
— Level of power density (fuel inventory) in the core: from ~500 $MW{\cdot}m^{-3}$ to ~250 $MW{\cdot}m^{-3}$;
— Fuel cycle duration: from 1–3 years to 5 years and more;
— Depth of fuel purification in the stage of its reprocessing: from 10^{-4} to 10^{-8};
— Expediency and method of minor actinide transmutation;
— Maximum possible fuel burnup or optimization of the fuel burnup value.

2.1. Conceptual ways of reactor safety improvement

The following main ways of safety improvement for the new generation reactors are considered:

— Minimization of excess reactivity for the fuel burnup;
— Decrease of sodium void reactivity effect;
— Use of passive devices for reactivity control;
— Use of passive devices for decay heat removal.

In general, the block of requirements is aimed at the development of internal self-protection equipment.

Concept of zero excess reactivity for the fuel burn-up (equilibrium fuel). This concept is one of the most often used in the Russian approaches. It implies the requirement for decreasing excess reactivity of the core down to the value assuring elimination of reactor excursion on prompt neutrons by only controlling core characteristics without the use of reactivity control equipment (control and safety rods). This concept can be implemented most effectively if high density fuel (nitride fuel) is used (see Fig. 1). This results, in particular, in the unique stability of characteristics of the BN-1200 reactor core.

Analysis of this concept led to the necessity of assurance of very high accuracy in forecasting excess reactivity value under conditions of calculation errors and technological uncertainties concerning the amount and composition of the fuel. In many cases, they cast doubt on the possibility of meeting such requirements. However, analysis of the experiments carried out by the authors in the BN-600 showed the possibility, in principle, of forecasting excess reactivity value to an accuracy of ~0.2 Δk/k by means of sharing calculation and experimental data [9, 10] (see Fig. 2).

Concept of zero sodium void reactivity effect. Before 2009, in the Russian regulatory documents limitations were imposed on density reactivity effect, which led Russian specialists to the ‘zero SVRE’ concept. This implies formation of the core design with sodium plenum assuring an integral void reactivity effect close to zero.

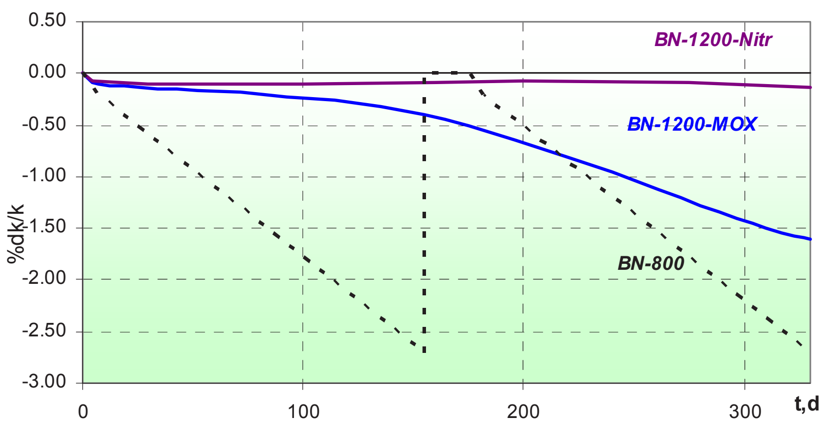


FIG. 1. Excess reactivity behaviour in reactors with high power density (BN-800) and in reactors with low power density (BN-1200) [8].

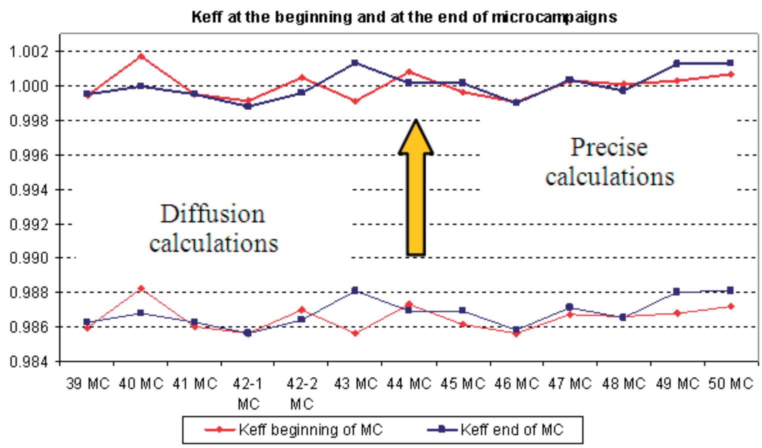


FIG. 2. Evaluation of the accuracy of forecast criticality of the BN-600 reactor [9].

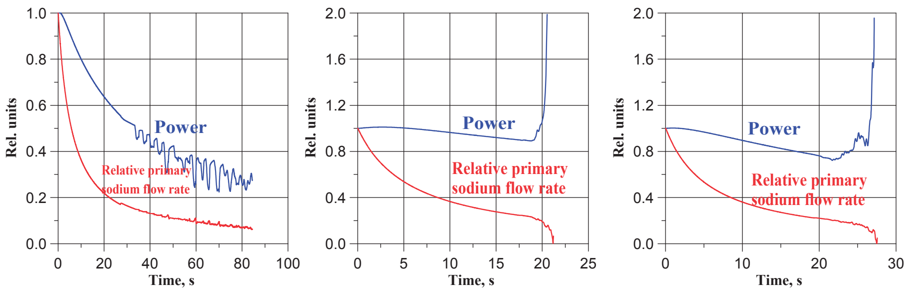


FIG. 3. Characteristics of ULOF accident in BN-1200 reactor (with sodium plenum (left), without sodium plenum (middle), with increased height of core (right))[10].

Now, in spite of modifications made to the regulatory documents to admit a positive SVRE value, the main design approaches based on SVRE decrease by the introduction of sodium plenum and core height decrease down to 80–85 cm (in order to eliminate core meltdown in the case of ULOF accident, see Fig. 3) are still kept for the advanced fast reactors [11].

2.2. Requirements to the reactor for closed nuclear fuel cycle implementation

One of the most important conceptual advantages of fast reactors with a BR > 1 is the possibility of their operation in closed nuclear fuel cycle (CNFC), i.e. when uranium-238 only is consumed. In Fig. 4, presented is the basic way of switching from nuclear power on thermal neutron reactors with the open fuel cycle to CNFC in the Russian Federation. This implies the initial startup using Pu from spent nuclear fuel (SNF) from WWER reactors.

Calculations showed significant change of attitude towards the concept of doubling time minimization. In the early stage of fast reactors, introduction of the role of Pu produced by fast reactors would not be important for nuclear power (see Fig. 5) [12].

Role of fuel breeding. Nowadays, the role of the fuel breeding is not as highly appreciated as it was in connection with typical requirements to physical conceptions of the early generation fast reactors. According to various evaluations, a BR value ranging from 1.05 to 1.2–1.3 is considered acceptable.

Role of startup loading (core power density). High power density was one of the key elements of the fast reactor concept, determining, along with other solutions, the choice of sodium as coolant. The recent evaluation of asymptotic capacity of nuclear power plants, W_{NP}^{as}, from startup critical loading $M_{cr.}$

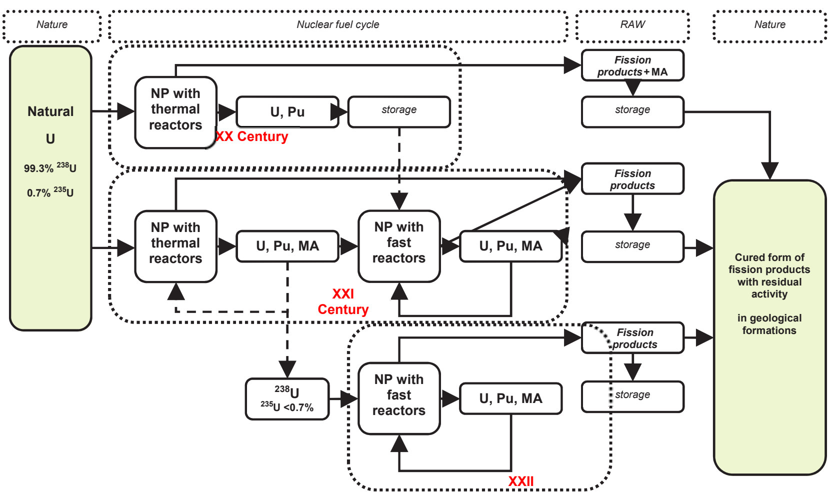


FIG. 4. Flowchart of changes from open to CNFC.

(see relationship below) shows a moderate effect of power density because of anti-correlation of $M_{cr.}$ and duration of reactor related fuel cycle section, T_R:

$$W_{NP}^{as} = \frac{M_{Pu}}{M_{cr}} \cdot \frac{1}{1 + T_{FC}/T_R}$$

This means the possibility of crucial abandoning of high power density of the core, which was specific for the early generation fast reactors.

Role of duration of external fuel cycle. The role of the external fuel cycle remains significant (see Fig. 6). Although the results of estimates of numerical characteristics made by various specialists are somewhat different, these estimates, as a rule, show appearance of a stagnation period in NP development caused by the lack of Pu. In order to decrease this risk, nuclear technology should be sensitized to the short fuel cycle (1–3 years) and, hence, operation in the cycle with hot fuel.

Concept of fast reactor start with uranium-235. It is known that fast reactors are also capable of operating with uranium-235. It was enriched fuel that was used in the BN-350 reactor and it is used now in the BN-600 reactor.

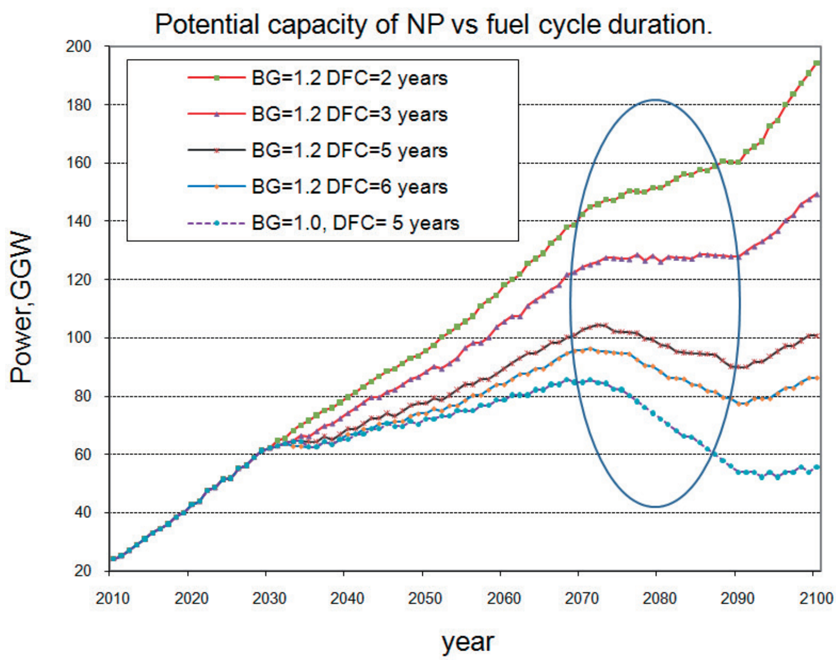


FIG. 5. Role of different type reactors in SNF accumulation.

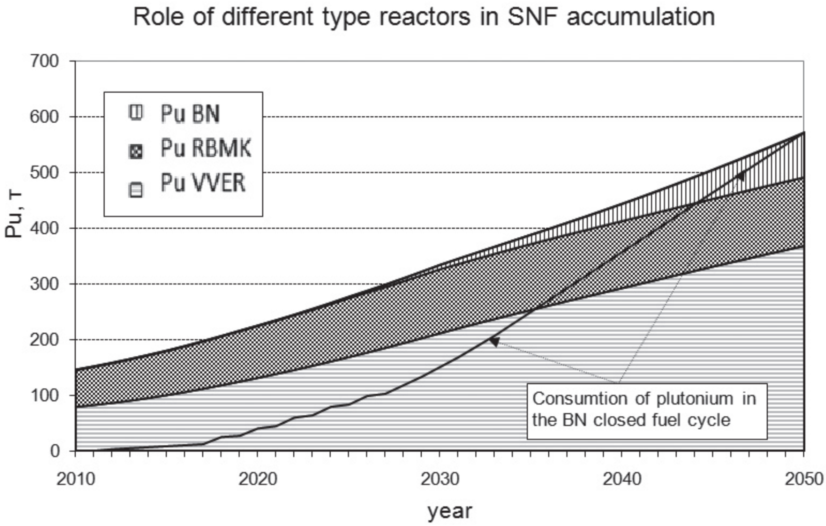


FIG. 6. Potential capacity of NP versus fuel cycle duration.

Recently, the new concept of fast reactor start has been studied extensively in the Russian Federation. This concept, initiated by V.V. Orlov, is called ‘concept of uranium start’. It implies the use of enriched uranium in startup loadings of fast reactors in order to breed plutonium and change gradually for mixed fuel.

This concept makes it possible to drastically eliminate the limitations on NP capacity on condition of availability of the unspent uranium stockpile for the start. Owing to better neutron balance in fast reactors, the use of uranium fuel in this reactor would result in considerable Pu amount advantage as compared to its use in thermal reactors.

This concept is now considered in connection with the BREST reactor, and preliminary estimates have been made as applied to the BN-1200 [13]. Evaluations show the feasibility of gradual change from uranium used for startup to plutonium during 10–15 years. There is one very complicated technical problem that has not been solved, namely, assurance of low excess reactivity for the fuel burnup during the whole transient period. Certainly, this concept contains many questionable and unexamined points, so this problem requires special attention.

2.3. Minor actinide transmutation

The concept of closed nuclear fuel cycle solves automatically the problem of management of SNF, but it gives rise to the problem of management of radioactive waste (RAW) resulting from SNF reprocessing. It is well known that the group of minor actinides is most critical in this sense. Experimental studies on this problem were carried out in the BN-350 reactor by the pin-type witness specimen method [14].

Analysis of methods of minor actinide destruction, taking into account results of studies carried out in the BN-350 (see Fig. 7), led the authors to the following conclusions:

— The effectiveness of minor actinide incineration and their transformation into fission products in power reactors is not too high because of parasitic neutron capture.
— Approximately two-thirds of reactions lead to the formation of secondary actinides (see Table 1) and the appearance of secondary activity, which sometimes would exceed initial activity (see Fig. 8).
— Minor actinide transformation into the basic fuel isotopes (similar to the fuel breeding process) through curium isotopes should be considered as the main method of their transmutation (see Fig. 9).
— Irradiation of short lived curium isotopes does not make sense and, therefore, these should be separated and stored until their decay to plutonium.
— It is inexpedient to carry out minor actinide transmutation separately from the basic fuel circulating in the CNFC, and this transmutation is practically senseless if CNFC is not adopted.
— Homogeneous transmutation along with fuel is considered as the most effective approach.
— Although heterogeneous transmutation is not excluded, it is, however, effective, if only recycling of fuel with minor actinides is synchronized with the basic fuel recycling.

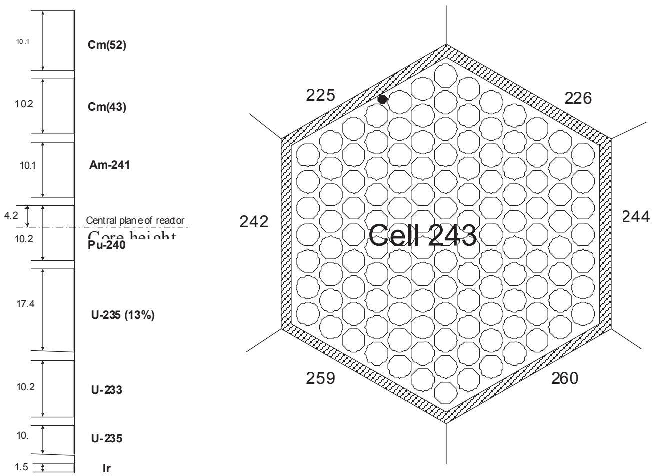


FIG. 7. Ampoule experiments in the BN-350.

TABLE 1. EVALUATED CHANGE OF MINOR ACTINIDE COMPOSITION DURING ONE IRRADIATION CYCLE

Nuclides	Burnup of basic isotope	Accumulation of secondary actinides	Actinide burnup
Am241	35%	25.5%	9.5%
Np237	35%	25%	10%
Cm244	44%	33%	10%
Pu240	20%	9%	11%
Pu238	34%	12%	22%

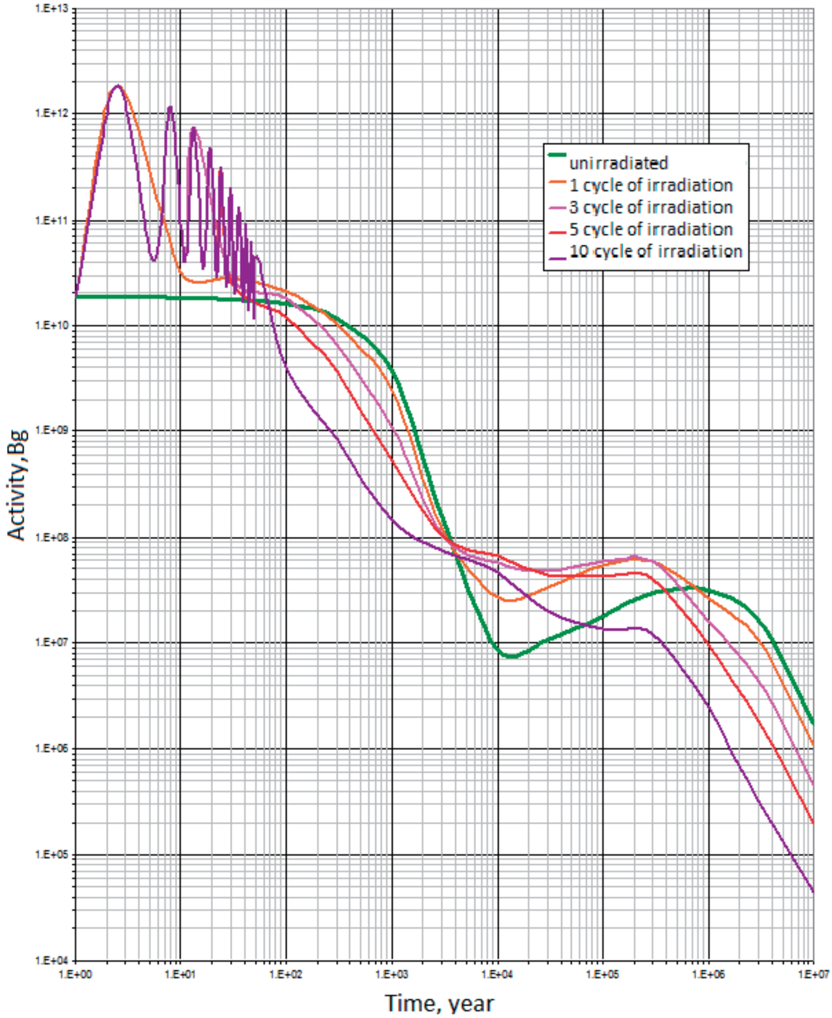


FIG. 8. Evaluated Am activity behaviour under multiple recycling conditions.

3. BASIC CONCEPTUAL DECISIONS ON ADVANCED REACTORS WITH SODIUM AND HEAVY LIQUID METAL COOLANTS

Construction of the BN-800, the first post-Soviet fast reactor in the Russian Federation (see Fig. 10) is now near completion. This project is aimed at the development of the fuel cycle infrastructure and mastering of the new types of fuel.

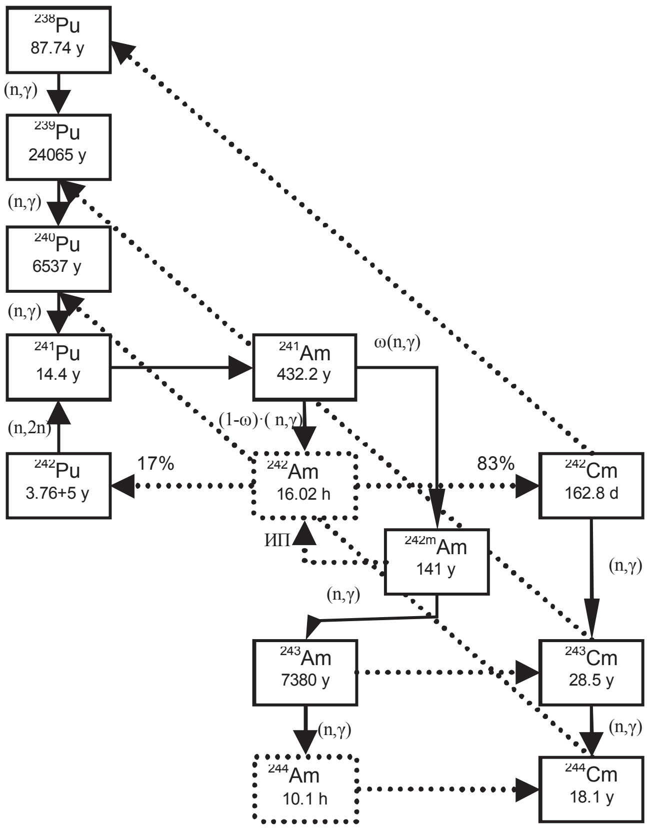


FIG. 9. Nuclide transmutation transitions caused by irradiation of Am-241.

The sodium plenum, which makes it possible to assure the zero void reactivity effect, as well as passive safety systems, are special features of BN-800 reactor design. Tests of these elements would lead to the progress in the area of fast reactor safety.

Two fast reactor designs, namely, BN-1200 with sodium coolant and BREST-1200 with lead coolant, pretend to the role of 4th generation reactors. Another area of studies on fast reactors concerns reactors with lead–bismuth coolant (SVBR-100/75). By now, this technology is considered exclusively in connection with small size reactors.

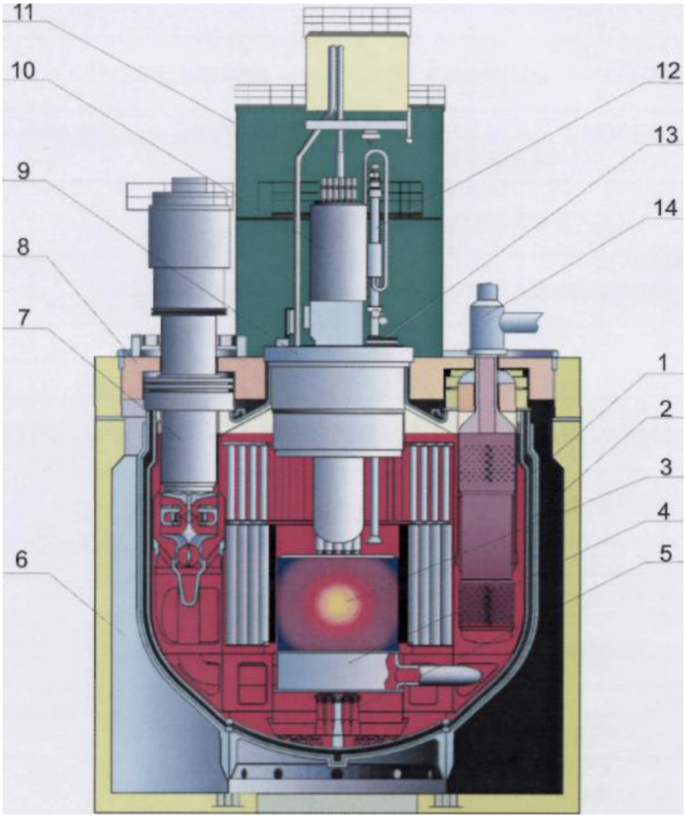


FIG. 10. BN-800 reactor (1 — reactor vessel, 2 — guard vessel, 3 — reactor core, 4 — core diagrid, 5 — core catcher, 6 — silo, 7 — main sodium pump, 8 — upper stationary shielding, 9 — large rotating plug, 10 –central rotating plug, 11 — protection cap, 12 — refueling mechanism, 13 — small rotating plug, 14 — intermediate heat exchanger).

Conceptual solutions adopted in BN-1200 design [15]:

— Basic fuel — mixed nitride;
— Low power density in the core;
— Fuel cycle duration — 3 years;
— Fuel breeding level — 1.2–1.3 with BR core ~1;
— Flattened core, sodium plenum is used for SVRE minimization;
— Minor actinide utilization in the basic fuel or heterogeneous fuel subassemblies is admitted;
— Integration of all primary sodium systems in the reactor vessel to eliminate radioactive sodium leaks.

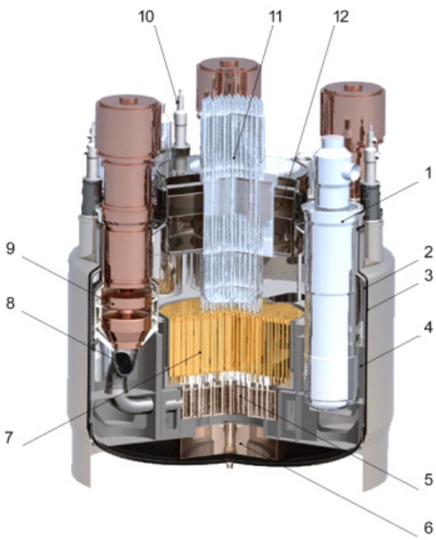


FIG. 11. BN-1200 reactor (1 — intermediate heat exchanger, 2 — reactor vessel, 3 — guard vessel, 4 — silo, 5 — core diagrid, 6 — core catcher, 7 — reactor core, 8 — pump nozzle, 9 —main sodium pump, 10 — cold trap, 11 — control rod drives, 12 — rotating plug).

Designers of the BN-1200 reactor (see Fig. 11) make it a point to significantly improve the reactor's economic characteristics by optimization of layout approaches, increase of load factor by changing to one-year refuelling interval and increase of fuel burnup.

The reactor plant of the BREST type with lead coolant is principally a new fast neutron reactor design of the Russian Federation. In the early stages of assimilation of the new type of coolant, creation of a prototype demonstration reactor BREST-OD-300 (see Fig. 12) [16] is assumed, in which the following conceptual solutions are implemented:

— Basic fuel — mixed nitride;
— Low power density of the core;
— Duration of external fuel cycle — 2 years (1 year duration without storing in IVS is also considered);
— Fuel breeding level: BR = BR core ~1.05 without blankets;
— Excess reactivity for the fuel burnup does not exceed effective fraction of delay neutrons;
— Elimination of the most severe accidents by means of natural properties specific for the reactor, its fuel, coolant, as well as reactor design facilitating implementation of these properties;

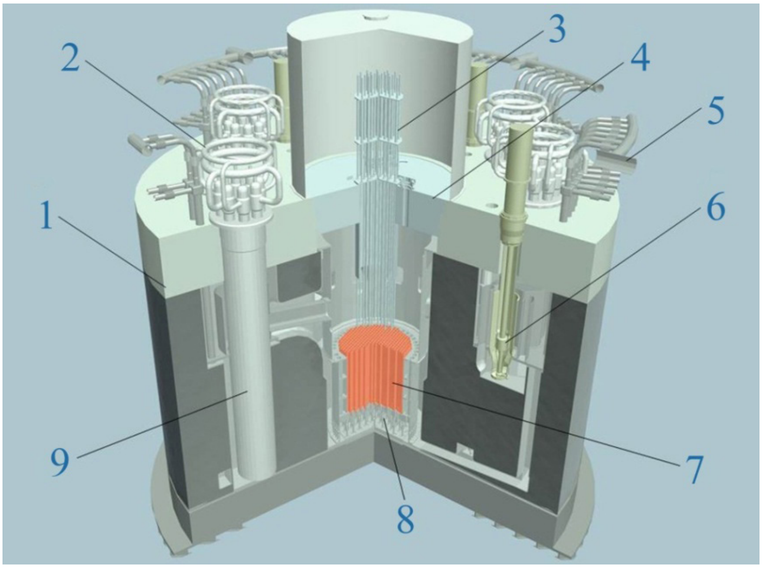


FIG. 12. BREST-OD-300 reactor. 1 — reactor vessel; 2 — steam-water collectors; 3 — control rod drives; 4 — rotating plug; 5 — channels of decay heat cooling; 6 - main pump; 7 — reactor core; 8 — core diagrid; 9 — steam generator.

— Operation in CNFC with rough cleaning of fuel from fission products and curium;
— Potential possibility of using enriched uranium nitride as starting fuel with further change for mixed fuel is under study.

Fundamentally, a new solution is to use non-burning coolant with a high boiling point, which gives additional advantages over sodium cooled reactors in the event of a reliable development of new coolant technologies. The use of non-burning coolant with a high boiling point is a brand new solution giving additional advantages compared to sodium cooled reactors, on condition of reliable mastering of the new coolant technology.

4. CONCLUSIONS

For both Russian specialists and those from countries where nuclear power is under development, there is no doubt in the necessity to change the new types of reactor, namely, fast neutron reactors. Certainly, the time required for this change is to be determined.

Russian experience in developing fast reactors has proved clearly the scientific justification of conceptual physical principles and their technical feasibility. However, the potential for fast reactors due to their physical features has not been realized to its full extent up to now. In order to assure the real possibility of transition to the nuclear power with fast reactors by about 2030, it is necessary to consistently update fast reactor designs for solving the following key problems:

— Increase in self-protection level of reactor core;
— Improvement of technical and economic characteristics;
— Solution of the problems related to the fuel supply of nuclear power and assimilation of the CNFC;
— Disposal of long lived radioactive waste and transmutation of minor actinides.

The Russian programme on the development of the basic concepts of the new generation reactors implies successive solution of the above problems, and development and assimilation of the new reactors, namely, the BN-800 and BN-1200 reactors with sodium coolant and the BREST type reactor with heavy liquid metal coolant.

ACKNOWLEDGEMENT

The authors express their gratitude to V.I. Rachkov, V.M. Poplavsky, B.A. Vasiliev, Yu.M. Ashurko, Yu.E. Shvetsov, V.V. Orlov and V.S. Smirnov for fruitful discussion of the topics considered in the paper and provision of materials for consolidation in this paper.

REFERENCES

[1] KHOMYAKOV, Yu.S., VASILIEV, B.A., LOPATKIN, A.V., et al., “Substantiation of physical concepts of fast reactors in Russia: experience and prospects”, paper presented at Int. Conf. Advances in Reactor Physics (PHYSOR), Interlaken, Switzerland, 2008.

[2] LEYPUNSKY, A.I., et al., “Experimental fast reactors in the USSR”, Proc. 2nd Int. Conf. on Peaceful Use of Atomic Energy, Geneva, 1958, Moscow, Atomizdat (1959) R/2129.

[3] LEYPUNSKY, A.I., et al., Experimental fast neutron reactor, Atomnaya Energiya, **2** (1957) 497–501.

[4] ORLOV, V.V., et al., Studies on physical characteristics in the stage of the BN-350 reactor start-up, Atomnaya Energiya, **36** 2 (1974).

[5] KAZANSKY, Yu.A., et al., Studies on physical characteristics of the BN-600 reactor, Atomnaya Energiya **55** 1 (1983).

[6] VASILIEV, B.A., KUZAVKOV, N.G., KHOMYAKOV, Yu.S., et al., Experience and prospects of updating of the BN-600 reactor core, Bulletin of Higher Education Institutions, Nuclear Power 1 (2011).

[7] KHOMYAKOV, Yu.S., POPLAVSKY, V.M., MATVEEV, V.I., et al., "Core design and fuel cycle of advanced fast reactor with sodium coolant", Proc. Int. Conf. on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), Kyoto, Japan, 2009, IAEA, Vienna (2012).

[8] KHOMYAKOV, Yu.S., ZHELTYSHEV, V.A., SELEZNYOV, E.F., BELOV, S.B., "Analysis of accuracy of reactivity balance components in BN-600 power fast reactor", paper presented at Int. Conf. Advances in Reactor Physics (PHYSOR), Pittsburgh, PA, 2010.

[9] MOISEEV, A.V., SEMENOV, M.Yu., KHOMYAKOV, Yu.S., et al., "Experimental and analytical justification of reactivity balance and power profile of the BN-600 reactor core", Bulletin of Higher Education Institutions, Nuclear Power 1 (2011).

[10] POPLAVSKY, V.M., MATVEEV, V.I., KHOMYAKOV, Yu.S., et al., Studies on influence of sodium void reactivity effect on the concept of the core and safety of advanced fast reactor, J. Nucl. Sci. Technol. **48** 4 (2011).

[11] MATVEEV, V.I., KHOMYAKOV, Y.S., Technical physics of fast reactors with sodium coolant, Workbook for Higher Education Institutions, (RACHKOV, V.I., Ed) MEI Publishing House, Moscow (2012).

[12] ANDREEVA, K.A., MALYSHEVA, I.V., MATVEEV, V.I., "Calculations of physical characteristics of the fuel cycle of BN type reactors with initial uranium loading and further change for breeding mode", Proc. Neitronika Workshop on Neutron Physics Issues of Nuclear Power (SSC RF–IPPE), Vol. 1, Obninsk, Russian Federation (2012) 52–60.

[13] KHOMYAKOV, Yu.S., KOCHETKOV, A.L., SEMENOV, M.Yu., et al., "Evaluation of the activation and burn-up experiments carried out in BN-350 reactor", paper presented at Int. Conf. Advances in Reactor Physics (PHYSOR), Seoul, Korea, 2002.

[14] KOTCHETKOV, A.L., NEROZIN, N.A. , KHOMYAKOV, Y.S., et al., "Calculation and experimental studies on minor actinides samples irradiations in fast reactors", paper presented at Int. Conf. Transmutation and Partitioning (P&T-2002) Jiju, Korea, 2002.

[15] VASILIEV, B.A., et al., Large size advanced fast reactor with sodium coolant, Bulletin of Higher Education Institutions, Nuclear Power 1 (2011).

[16] SMIRNOV, V.S., BREST-OD – prototype of inherently safe fast reactor, Atomnaya Energiya **113** 1 (2012).

FUTURE ADVANCED NUCLEAR SYSTEMS AND THE ROLE OF MYRRHA AS A WASTE TRANSMUTATION R&D FACILITY

H.A. ABDERRAHIM
Belgian Nuclear Research Centre (SCK•CEN),
Mol, Belgium

Abstract

MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications) is an experimental accelerator driven system (ADS) currently being developed at SCK•CEN in replacement of its material testing reactor BR2. The MYRRHA facility is conceived as a flexible fast spectrum irradiation facility, which is able to run in both subcritical and critical modes. The applications catalogue of MYRRHA includes fuel developments for innovative reactor systems, material developments for GEN IV systems and fusion reactors, doped silicon production, radioisotope production and fundamental science applications thanks to the high power proton accelerator. Next to these applications, MYRRHA will demonstrate the ADS full concept by coupling a high power proton accelerator, a multi-megawatt spallation target and a subcritical reactor at reasonable power level to allow operational feedback, scalable to an industrial demonstrator and allow the study of efficient transmutation of high level nuclear waste. Since MYRRHA is based on heavy liquid metal technology, namely lead–bismuth eutectic, it will be able to significantly contribute to the development of lead fast reactor (LFR) technology and will fill the role of the European Technology Pilot Plant in the roadmap for LFR. The current design of the MYRRHA ADS and its ability to contribute to the European Commission strategy for high level waste management through partitioning and transmutation are discussed in this paper.

1. INTRODUCTION

One of the major challenges for our society is the increasing demand for energy in general and electricity in particular. During the last century, our energy supply was mainly based on fossil fuels. Nowadays, we are confronted with decreasing classical hydrocarbon reserves and excessive CO_2 emissions. At the same time, renewable energy sources cannot satisfy the complete demand. For these reasons, it is clear that nuclear energy needs to be part of the energy basket of the future.

Nuclear energy nowadays is generated by means of thermal reactors. The use of thermal neutrons has a negative effect on the efficient use of uranium. Indeed, thermal neutrons can fission only ^{235}U, which is present at 0.7% in natural uranium. The major part of the ore, ^{238}U, remains unusable for fission in reactors of the current generation with a thermal neutron spectrum. If nothing changes from a technological point of view and in the case of a rapid development of nuclear energy, known uranium resources will become scarce before the end of the century. Reactors with a fast neutron spectrum allow using the remaining 99.3% of mineral uranium as fuel by transforming ^{238}U in ^{239}Pu. As a result, the present uranium resources can be used more efficiently, leading to uranium resources for more than several thousands of years.

The generation of nuclear energy from uranium produces, besides energy, high level nuclear waste (HLW). For this HLW, a technical and socially acceptable solution is necessary. The time scale needed for the radiotoxicity of the spent fuel to drop to the level of natural uranium is very long (i.e. of the order of 500 000 to 1 million years) (see Fig. 1). The present proposed solution for the spent fuel or the HLW resulting from classical reprocessing consists of burying it in geological storage.

The concept of partitioning and transmutation (P&T) has three main goals: reduce the radiological hazard associated with spent fuel by reducing the inventory of minor actinides, reduce the time interval required to reach the radiotoxicity of natural uranium and reduce the heat load of the HLW packages to be stored in the geological disposal, hence reducing the footprint of the geological disposal.

Advanced management of HLW through P&T consists in advanced separation of the minor actinides (americium, curium and neptunium) and some fission products of high level radioactivity with a long half-life present in the nuclear waste and their transmutation in dedicated burners to reduce the radiological and heat loads on the geological disposal. The time scale needed for the radiotoxicity of the waste to drop to the level of natural uranium will be reduced from a 'geological' value (500 000 to 1 million years) to a value that is comparable to that of human activities (several hundreds of years) [1–3]. Transmutation of the minor actinides is achieved through fission reactions and therefore fast neutrons are needed in the dedicated burners.

Indeed, the fission cross-sections of the considered minor actinides have a high energy threshold above 0.75 MeV. Another characteristic of Am, Cm and Np is their smaller values of the delayed neutron fraction (ranging between 0.1 and 0.03%) compared to that of ^{235}U (0.7%) or ^{238}U (1.5%) or even of ^{239}Pu (0.4%) making a critical fast reactor burner loaded with a large fraction of the core with minor actinides much more difficult to control. Therefore, if one is aiming at an efficient transmutation of minor actinides then the subcriticality of the core is not a luxury but mandatory. This is the reason why accelerator subcritical driven systems are studied for minor actinides efficient and rapid transmutation.

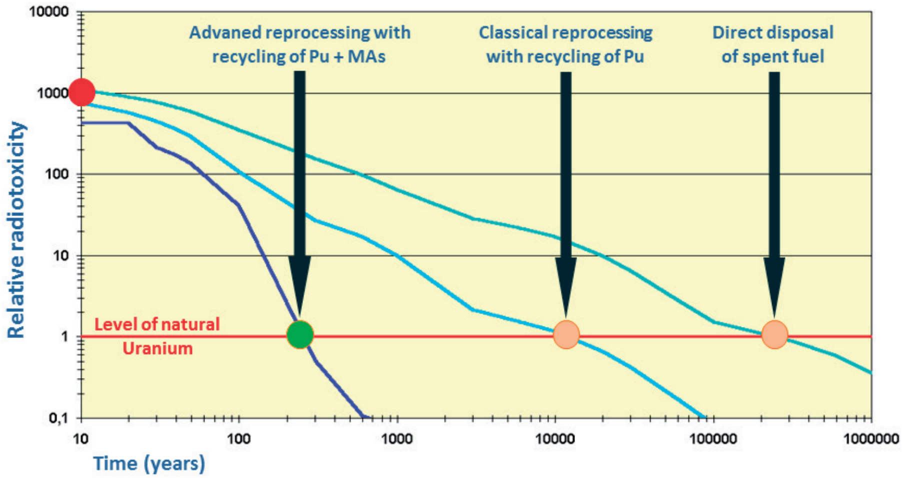


FIG. 1. Radiotoxicity of radioactive waste.

2. THE CURRENT DESIGN OF THE MYRRHA ADS

MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications) is the flexible experimental accelerator driven system (ADS) in development at SCK•CEN. MYRRHA is able to work both in subcritical (ADS) and in critical modes. In this way, MYRRHA should target the following applications catalogue:

— To demonstrate the full ADS concept by coupling the three components (accelerator, spallation target and subcritical reactor) at reasonable power level to allow operation feedback and scalability to an industrial demonstrator.

— To allow the study of the efficient technological transmutation of high level nuclear waste, in particular minor actinides that would require high fast flux intensity ($\Phi_{>0.75MeV} = 10^{15}$ n/cm^2·s).

— To be operated as a flexible fast spectrum irradiation facility allowing for:
  - Fuel developments for innovative reactor systems, which need irradiation rigs with a representative flux spectrum, a representative irradiation temperature and high total flux levels ($\Phi_{tot} = 5 \times 10^{14}$ to 10^{15} n·cm^{-2}·s); the main target will be GEN IV systems which require fast spectrum conditions.

- Material developments for GEN IV systems, which need large irradiation volumes (3000 cm³) with high uniform fast flux level ($\Phi_{>1\ \mathrm{MeV}}$ = ~ 5 × 10^{14} n·cm^{-2}·s) in various irradiation positions, representative irradiation temperature and representative neutron spectrum conditions; the main target will be fast spectrum GEN IV systems.
- Material developments for fusion reactors which need also large irradiation volumes (3000 cm³) with high fast flux level ($\Phi_{>1\ \mathrm{MeV}}$ = ~ 5 × 10^{14} n·cm^{-2}·s) with low gradients, a representative and controlled irradiation temperature and a representative ratio appm He/dpa(Fe)[1] = 10.
- Radioisotope production for medical and industrial applications by:
  - Holding a backup role for classical medical radioisotopes;
  - Focusing on R&D and production of radioisotopes requesting very high thermal flux levels (Φ_{thermal} = 2 to 3 × 10^{15} n·cm^{-2}·s) due to double capture reactions.
- Industrial applications, such as Si-doping that needs a thermal flux level depending on the desired irradiation time. For a flux level Φ_{thermal} = 10^{13} n·cm^{-2}·s, an irradiation time of the order of days is needed and for a flux level of Φ_{thermal} = 10^{14} n·cm^{-2}·s, an irradiation time of the order of hours is needed to obtain the required specifications.

MYRRHA started from the ADONIS project (1995–1997), which was the first project at SCK•CEN where the coupling between an accelerator, a spallation target and a subcritical core was studied. ADONIS was a small irradiation facility, having the production of ^{99}Mo as its single objective. In 1998, the ad-hoc scientific advisory committee recommended extending the purpose of the ADONIS machine to become a material testing reactor for material and fuel research, to study the feasibility of transmutation of minor actinides and to demonstrate the principle of the ADS at a reasonable power scale. Since 1998, the project is has been called MYRRHA.

In 2005, MYRRHA consisted of a proton accelerator delivering 350 MeV at 5 mA to a windowless spallation target coupled to a subcritical fast core of 50 MW(th). This 2005 version is the 'MYRRHA — draft 2' design [4]. This 2005 design was used as a starting base within the FP6 EUROTRANS integrated project [5], which resulted in the XT-ADS [6] (Experimental Demonstration of the Technical Feasibility of Transmutation in an Accelerator Driven System) design, where a linear proton accelerator delivers a 600 MeV at 3.2 mA beam into the spallation target. The reactor power of XT-ADS was 57 MW(th).

---

[1] appm He/dpa = atomic parts per million helium per displacement per atom.

The XT-ADS design was taken as a starting point for the work performed in the FP7 CDT project [7], which resulted in the MYRRHA-FASTEF (MYRRHA Fast Spectrum Transmutation Experimental Facility) design. The current design of MYRRHA-FASTEF is described in this paper.

2.1. Accelerator

The accelerator is the driver of MYRRHA since it provides the high energy protons that are used in the spallation target to create primary neutrons which in turn feed the subcritical core. In the current design of MYRRHA, the machine must be able to provide a proton beam with an energy of 600 MeV and an average beam current of 3.2 mA. The beam is delivered in continuous wave mode. Once a second, the beam is shut off for 200 μs so that accurate on-line measurements and monitoring of the subcriticality of the reactor can take place. The beam is delivered to the core from above through a beam window.

Accelerator reliability is a crucial issue for the operation of an ADS. A high reliability is expressed by a long mean time between failure (MTBF), which is commonly obtained by a combination of over-design and redundancy. On top of these two strategies, fault tolerance in the high energy section of the linac (above 17 MeV) must be implemented to obtain the required MTBF. Fault tolerance will allow the accelerator to recover the beam within a beam trip duration tolerance after failure of a single cavity. In the MYRRHA case, the beam trip duration tolerance is 3 seconds. Within an operational period of MYRRHA of 3 months, the number of allowed beam trips exceeding 3 seconds must remain under 10, shorter beam trips are allowed without limitations. The combination of redundancy and fault tolerance should allow obtaining a MTBF value in excess of 250 hours to meet the required number of beam trips per operation cycle of 3 months.

At present, proton accelerators with megawatt level beam power in continuous wave mode only exist in two basic concepts: sector focused cyclotrons and linear accelerators (linacs). Cyclotrons are an attractive option with respect to construction costs, but they do not have any modularity, which means that a fault tolerance scheme cannot be implemented. Also, an upgrade of its beam energy is not a realistic option. A linac, especially if made superconducting, has the potential for implementing a fault tolerance scheme and offers a high modularity, resulting in the possibility to recover the beam within a short time and increasing the beam energy.

A basic layout of the MYRRHA accelerator, aiming at maximizing its efficiency, its reliability (or MTBF) and its modularity, is provided in Fig. 2.

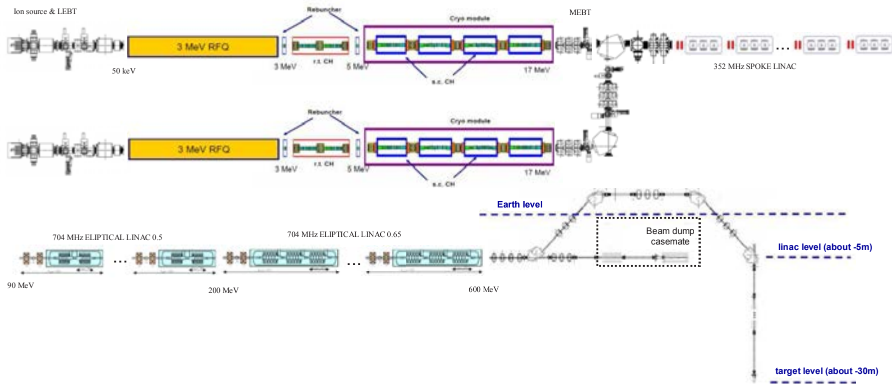


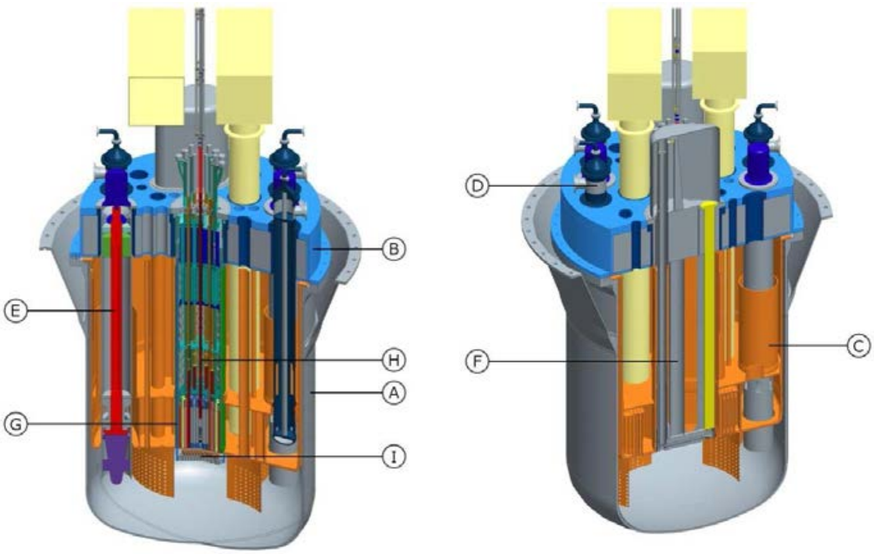
FIG. 2. A schematic layout of the reference design of the MYRRHA accelerator.

2.2. Core and primary system

The main components/systems of the current MYRRHA-FASTEF design are of the same MYRRHA/XT-ADS type, as defined within the EUROTRANS project, with only increased size. The primary and secondary systems have been designed to evacuate a maximum core power of 100 MW_{th}. All the MYRRHA-FASTEF components are optimized for the extensive use of the remote handling system during component replacement, inspection and handling.

Since MYRRHA-FASTEF is a pool type ADS, the reactor vessel houses all the primary systems. In previous designs of MYRRHA, an outer vessel served as secondary containment in case the reactor vessel leaked or broke. In the current design, the reactor pit fulfils this function, improving the capabilities of the reactor vault air cooling system. The vessel is closed by the reactor cover which supports all the in-vessel components. A diaphragm inside the vessel acts to separate the hot and cold LBE plenums, to support the in-vessel fuel storage and to provide a pressure separation. The core is held in place by the core support structure consisting of a core barrel and a core support plate. Figure 3 shows vertical sections of the MYRRHA-FASTEF reactor showing its main internal components.

At the present state of the design, the reactor core (Fig. 4) consists of mixed oxide fuel pins, typical for fast reactors. A major change with respect to the previous version of the core is the switch from a windowless loop type spallation target to a window beam tube type spallation target. The previous version needed three central hexagons to house the spallation target while the present day design only needs one central hexagon. To better accommodate this central target, the fuel assemblies' size is slightly increased as compared to the



A. Reactor Vessel
B. Reactor Cover
C. Diaphragm
D. Primary Heat Exchanger
E. Primary Pump
F. In-vessel Fuel Handling Machine
G. Core
H. Above Core Structure
I. Core Restraint System

FIG. 3. Section of the MYRRHA-FASTEF reactor.

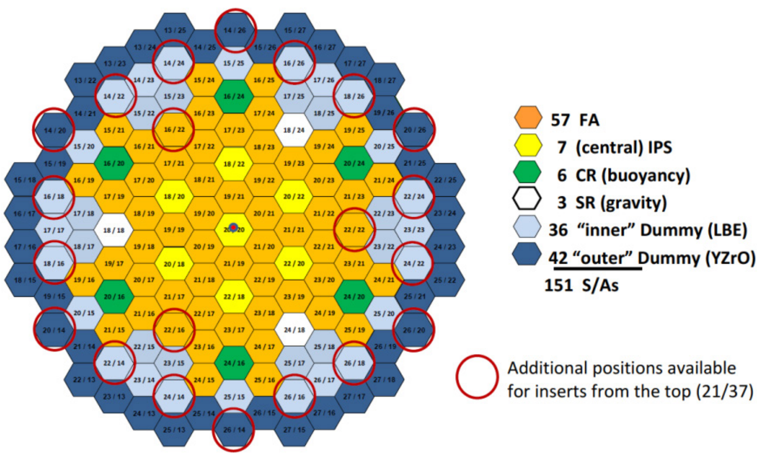


FIG. 4. Cross-section of the MYRRHA-FASTEF core, showing the central target, the different types of fuel assembly and dummy components.

MYRRHA/XT-ADS design. Consequently, the in-pile test sections (IPS), which will be located in dedicated FAs positions, are larger in diameter, giving more flexibility for experiments. Thirty seven positions can be occupied by IPSs or by the spallation target (the central one of the core in subcritical configuration) or by control and shutdown rods (in the core critical configuration). This gives a large flexibility in the choice of the more suitable position (neutron flux) for each experiment.

The requested high fast flux intensity has been obtained by optimizing the core configuration geometry (fuel rod diameter and pitch) and maximizing the power density. We will be using, for the first core loadings, 15-15Ti stabilized stainles steel as cladding material instead of T91 ferritic-martensitic steel that will be qualified progressively further on during MYRRHA operation for later use. The use of LBE as coolant permits lowering the core inlet operating temperature (down to 270°C), decreasing the risk of corrosion and allowing increase of the core ΔT. This, together with the adoption of reliable and passive shutdown systems, will permit meeting the high fast flux intensity target.

As depicted in Fig. 4, showing a critical core layout (with 7 central IPS) at the equilibrium of the fuel cycle, 37 positions are available for multi-functional channels that can host:

- Fuel assembly and dummy, loaded from the bottom (in all the 151 positions);
- IPS, control and scram rods, loaded from the top.

In subcritical mode, the accelerator (as described in the previous section) is the driver of the system. It provides the high energy protons that are used in the spallation target to create neutrons which in their turn feed the subcritical core. The accelerator is able to provide a proton beam with an energy of 600 MeV and a maximum current of 4 mA.

In subcritical mode, the spallation target assembly, located in the central position of the core, brings the proton beam via the beam tube into the central core region. The spallation heat deposit is dissipated to the reactor primary circuit. The spallation module guarantees the barrier between the reactor LBE and the reactor hall and ensures optimal conditions for the spallation reaction. The spallation module assembly is conceived as an IPS and is easily removable or replaceable.

Differently from the critical layout (Fig. 4), in ADS mode the six control rods (buoyancy driven in LBE) and the three scram rods (gravity driven in LBE) will be replaced by absorbing devices to be adopted only during refuelling. Thanks to the (aimed and attained) flexibility, such absorbing devices will be implemented by adopting the control rods, but they will be controlled manually only by the operator.

As in critical mode, two different kinds of dummy assemblies are foreseen (Fig. 4):

- An internal ring surrounding the fissile zone made of LBE dummy assemblies (to increase neutron ‘reflection’);
- An external ring made of dummy assemblies having the same structure of FA with clads filled with YZrO pellets (to shield the core barrel).

The primary, secondary and tertiary cooling systems have been designed to evacuate a maximum thermal core power of 110 MW. The 10 MW more than the nominal core power account for the power deposited by the protons, for the power of in-vessel fuel and for the power deposited in the structures by γ-heating. The average coolant temperature increase in the core in nominal conditions is 140°C with a coolant velocity of 2 m/s. The primary cooling system consists of two pumps and four primary heat exchangers (PHX).

The primary pumps will deliver the LBE to the core with a mass flow rate of 4750 kg/s (453 L/s per pump). The working pressure of the pump is 300 kPa. The pump will be fixed at the top of the reactor cover, which is supposed to be the only supporting and guiding element of the pump assembly.

The secondary cooling system is a water cooling system, while the tertiary system is an air cooling system. These systems function in active mode during normal operation and in passive mode in emergency conditions for decay heat removal.

The main thermal connection between the primary and secondary cooling systems is provided by the PHX (Fig. 5). These heat exchangers are shell and tube, single-pass and counter-current heat exchangers. Pressurized water at 200°C is used as secondary coolant, flowing through the feedwater pipe in the centre of the PHX to the lower dome. All the walls separating the LBE and water plena (feedwater tube, lower dome and upper annular space) are double walled to avoid pre-heating of the secondary coolant and to prevent water leaking in the LBE in case of tube rupture.

In case of loss of the primary flow (primary pumps failure), the PHXs are unable to extract the full heat power. In such cases, the beam must be shut off in the subcritical case and the shutdown rods inserted in the critical case. Decay heat removal (DHR) is achieved by natural convection. Ultimate DHR is done through the reactor vessel cooling system (RVACS, reactor vessel air cooling system) by natural convection.

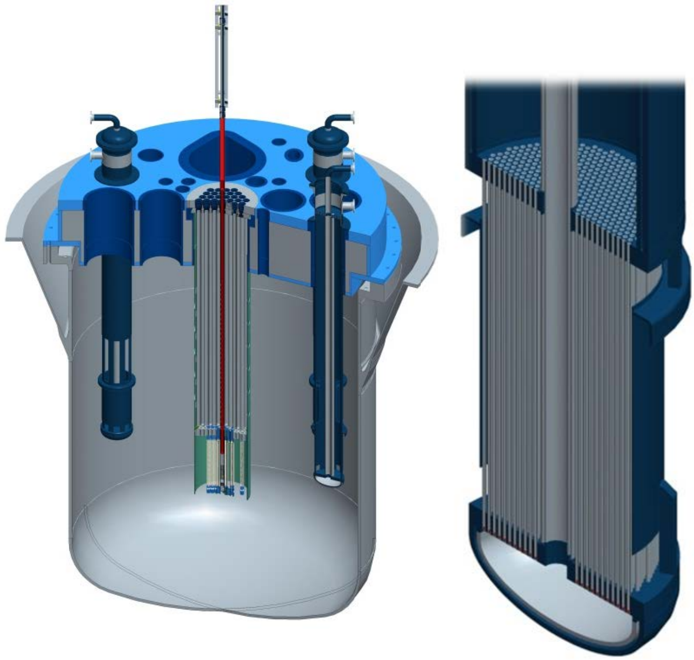


FIG. 5. Heat exchangers.

The interference of the core with the proton beam, the fact that the room located directly above the core will be occupied by significant instrumentation and IPS penetrations, together with core compactness, result in insufficient space for fuel handling to (un)load the core from above. Since the very first design of MYRRHA, fuel handling is performed from underneath the core. Fuel assemblies (FAs) are kept by buoyancy under the core support plate.

Two fuel handling machines are used, located at opposite sides of the core (Fig. 6). Each machine covers one side of the core. The use of two machines provides sufficient range to cover the necessary fuel storage positions without the need for an increase for the reactor vessel when only one fuel handling machine is used. Each machine is based on the well-known fast reactor technology of the ‘rotating plug’ concept using SCARA (selective compliant assembly robot arm) robots. To extract or insert the FAs, the robot arm can move up or down by about 2 m. A gripper and guide arm is used to handle the FAs: the gripper locks the FA and the guide has two functions, namely to hold the FA in the vertical orientation and to ensure neighbouring FAs are not disturbed when an FA is extracted from the core. An ultrasonic sensor is used to uniquely identify the FAs.

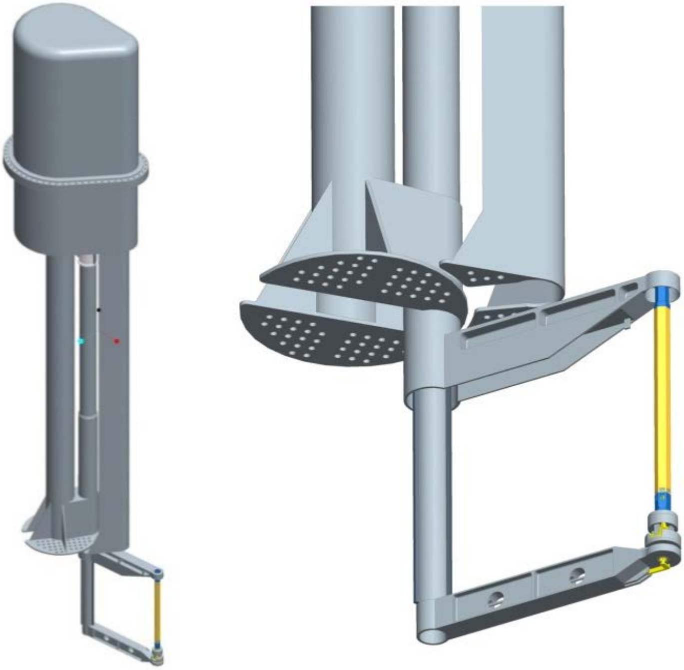


FIG. 6. The in-vessel fuel handling machine.

The in-vessel fuel handling machine will also perform in-vessel inspection and recovery of an unconstrained FA. Incremental single-point scanning of the diaphragm can be performed by an ultrasonic sensor mounted at the gripper of the IVFHM. The baffle under the diaphragm is crucial to the strategy as it limits the work area where inspection and recovery are needed. It also eliminates the need for additional recovery and inspection manipulators, prevents items from migrating into the space between the diaphragm and the reactor cover, and permits side scanning.

3. MYRRHA: A RESEARCH TOOL IN SUPPORT OF THE EUROPEAN ROADMAP FOR P&T

Spent nuclear fuel from LWRs contains a mixture of uranium and plutonium (up to 95% of the initial uranium mass), fission products and minor actinides such as neptunium, americium and curium. In the shorter term, the highly active but short lived fission products will dominate the activity of this spent fuel. However, the transuranics, including plutonium and the minor actinides (together with a few long lived fission products), are largely responsible for the long term radiotoxicity and heat production of LWR spent fuel.

The principle behind P&T is to isolate the minor actinides from this LWR spent fuel and transmute them. Since for these isotopes the fission to capture ratio increases with increasing neutron energy, a fast neutron spectrum facility is required. By burning the minor actinides, the long lived, heat producing component of spent fuel can be strongly reduced. This decreases the radiotoxicity of the spent fuel and its heat load. Both will ease the design and construction of a long term storage solution (geological disposal) from the engineering point of view.

P&T requires the development of an advanced fuel cycle. Currently, two major options for P&T are being studied worldwide: the single stratum approach where the minor actinides are burned in fast reactors that are deployed for electricity production and the double strata approach where the Pu is burned for electricity production in LWRs and fast reactors, whereas the minor actinides are burned in a dedicated facility (Fig. 7).

In the single stratum approach, the minor actinides can be mixed homogeneously in the fast reactor fuel or can be loaded in dedicated targets. In the homogeneous option, care must be taken in the analysis of the change in

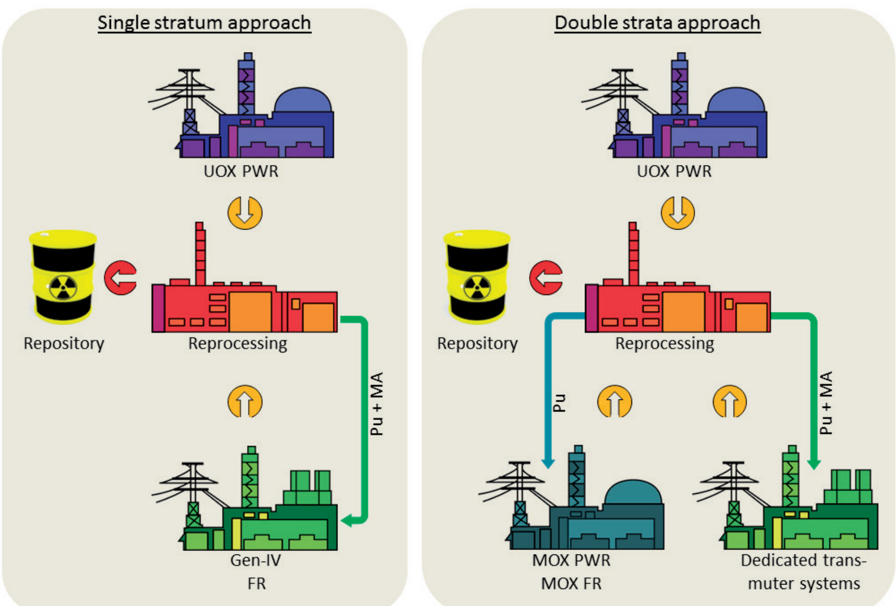


FIG. 7. Single versus double strata approach.

the core safety parameters such as delayed neutron fraction, Doppler constant, void coefficient. By increasing the concentration of minor actinides in the fuel mixture, these safety parameters typically go in the wrong direction and hence pose a threat to the reactor safety. Because of this, one expects a maximum of 4–5% of minor actinide loading in the fuel.

Also, the fabrication and reprocessing of this ‘spiked’ fast reactor fuel or the dedicated minor actinide target requires extra care since the presence of the minor actinides increases the heat production during these fabrication processes. The presence of Cm-244 will pose a shielding problem due its spontaneous fission and hence neutron emission.

Given the fact that only low amounts of minor actinides can be loaded per reactor, limited by a maximum concentration in the case of the homogeneous option or limited by the number of target positions in the heterogeneous option, a large number of reactors will be required to use this minor actinide spiked fuel or house these dedicated targets. This will certainly be the case when nations also decide to treat their legacy LWR waste and not only the minor actinides produced in this future advanced fuel cycle. This will imply a large number of transports of these fuels and targets from reprocessing site to fuel fabrication site and to transmutation sites and back.

In the double strata approach, a dedicated transmutation facility is foreseen in the form of an ADS. Because of the reactor physics properties of such an ADS (one does not rely on a subtle equilibrium such as a chain reaction, but the ADS subcritical core acts merely as a multiplier of a primary neutron source), one can devise fuels that have a very high minor actinide content. The EC-FP6 programme IP-EUROTRANS delivered the conceptual design of such an industrial transmuter (EFIT). In EFIT, 400 MW(th) core designs were made with uranium-free inert matrix fuels having a mixture of plutonium and minor actinides. In EFIT, the so-called 42-0 approach core was developed, meaning a core design which would be as plutonium neutral as possible (no burning or breeding of plutonium) and that could in optimal conditions burn 42 kg of minor actinides per TW·h power produced. This system was used in the EC-FP6 programme PATEROS which produced a roadmap for the development of P&T at the European level. The deployment of such an industrial transmuter such as EFIT would be very difficult for small nuclear countries and hence this scheme is optimal in a regional approach.

Since the burning of the minor actinides is done in a very concentrated manner, these industrial transmuters can be located near a fuel reprocessing and transmuter fuel fabrication facility, limiting transportation of hazardous materials. Calculations have indicated that the support ratio, i.e. the ratio of the total power of industrial transmuters to the total power of electricity generating systems, is about 6%. Also with this ‘concentrated’ approach, one can much easier envisage the burning of the LWR legacy waste in a reasonable amount of time without ‘bothering’ the regular electricity production installations.

Within the PATEROS project, a number of nuclear fuel cycle scenarios have been studied. Different regions have been identified: a group of countries that are stagnant with respect to nuclear energy production or in phase-out (‘Group A’, typically Belgium, Czech Republic, Germany, Spain, Sweden, Switzerland) and a group of countries that are developing an advanced fuel cycle with the deployment of fast reactors (‘Group B’, typically France). Different objectives were set concerning the burning of the minor actinides. Within the EC-F7 ARCAS project, which continues the work done in PATEROS, it was estimated that to burn the minor actinides present in Group A stockpiles in a reasonable time frame (less than 100 years), the group would need to deploy 7 EFIT-like facilities. If also Group B wanted to stabilize their minor actinide inventory, 15 EFIT-like installations would be needed and if total minor actinide elimination is required in Groups A and B, then 20 EFIT-like installations are to be built.

At the European level, a four building block strategy for P&T has been identified. Each block poses a serious challenge in R&D needed to reach an industrial scale deployment. These blocks are:

- Demonstration of advanced reprocessing of spent nuclear fuel from LWRs, separating uranium, plutonium and minor actinides;
- Demonstrate the capability to fabricate, at semi-industrial level, dedicated transmuter fuel heavily loaded in minor actinides;
- Design and construct one or more dedicated transmuters;
- Demonstration of advanced reprocessing of transmuter fuel together with the fabrication of new transmuter fuel.

MYRRHA will support this roadmap by playing the role of an ADS prototype (at reasonable power level) and as a flexible irradiation facility providing fast neutrons for the qualification of materials and fuel for an industrial transmuter. MYRRHA will be capable of irradiating samples of this inert matrix fuel but it is also foreseen to house fuel pins or even a limited number of fuel assemblies heavily loaded with minor actinides for irradiation and qualification purposes.

REFERENCES

[1] OECD, OECD Factbook 2011–2012: Economic, Environmental and Social Statistics, OECD Publishing, Paris (2012).

[2] OECD NUCLEAR ENERGY AGENCY, Physics and Safety of Transmutation Systems, A Status Report, OECD Publishing, Paris (2006).

[3] PATEROS, Partitioning and Transmutation European Roadmap for Sustainable Nuclear Energy, FI6W, Contract Number 036418.

[4] ABDERRAHIM, H.A., De BRUYN, D., et al., MYRRHA Project — Technical Description, SCK•CEN Rep. AND/HAA/DDB/3900.B043000/85/07-17bis (2007).

[5] EU Integrated Project EUROTRANS, contract nr. FI6W-CT-2004-516520.

[6] DE BRUYN, D., LARMIGNAT, S., WOAYE HUNE, A., MANSANI, L., RIMPAULT, G., ARTIOLI, C., "Accelerator driven systems for transmutation: main design achievements of the XT-ADS and EFIT systems within the FP6 IP-EUROTRANS integrated project", Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP), San Diego, California, 2010, American Nuclear Society, LaGrange Park, IL (2010).

[7] DE BRUYN, D., BAETEN, P., LARMIGNAT, S., WOAYE HUNE, A., MANSANI, L., "The FP7 Central Design Team Project: Towards a fast-spectrum transmutation experimental facility", Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP), San Diego, California, 2010, American Nuclear Society, LaGrange Park, IL (2010).

THE FRENCH FAST REACTOR PROGRAMME: INNOVATIONS IN SUPPORT OF HIGHER STANDARDS

F. GAUCHÉ
Commissariat à l'énergie atomique et aux énergies alternatives,
Saclay, France

Abstract

Deriving from the feedback of experience of former sodium cooled fast reactors, very high levels of requirements have been set for the ASTRID reactor (Advanced Sodium Technological Reactor for Industrial Demonstration). Innovations are needed to further enhance safety, reduce capital cost and improve efficiency, reliability and operability, making the Generation IV SFR an attractive option for electricity production. This puts great pressure not only on the R&D and design teams at CEA but also on the CEA's R&D and industrial partners. During the first phase of the ASTRID conceptual design (2010–2012), promising innovative options have been identified. They will be further developed in the next phases of ASTRID design studies, keeping a strong interaction between design and R&D.

1. INTRODUCTION

Fast neutron reactors have great potential as a sustainable energy source. In particular, sodium cooled fast reactors (SFRs) with a closed fuel cycle and potential for minor actinide burning allow improved use of natural resources and minimization of high level waste. Among the fast reactor systems, the SFR has the most comprehensive technological basis as result of the experience gained from decades of worldwide operation of several experimental, prototype and commercial size reactors. Currently, in France, a medium size (600 MW(e)) power demonstrator named ASTRID (Advanced Sodium Test Reactor for Industrial Demonstration) has been proposed and endorsed at the European level as the reference concept for Generation IV fast reactors.

Deriving from the feedback of experience, very high levels of requirements have been set for the ASTRID reactor. Innovations are needed to further enhance safety, reduce capital cost and improve efficiency, reliability and operability, making the Generation IV SFR an attractive option for electricity production. This puts great pressure not only on the R&D and design teams at CEA but also on the CEA's R&D and industrial partners. During the first phase of ASTRID conceptual design (2010–2012), promising innovation options have been identified. They will be further developed in the next phases of ASTRID design studies, keeping a strong interaction between design and R&D.

2. THE FRENCH FAST REACTOR PROGRAMME: ASTRID

France has a long history of R&D, construction and operation of fast neutron reactors and has developed over decades a very high level of competence and experience in this field. However, with the shutdown of the Superphenix reactor more than 10 years ago and of the Phenix reactor in 2009, we are now entering a new phase where the next generation of fast neutron reactors must be developed to prepare for their deployment in the future, since fast neutron reactors associated with their closed cycle are the only sustainable way for nuclear fission energy.

Fast neutron reactors have very interesting features in terms of sustainable energy:

- Excellent use of uranium resource and ability to recycle plutonium without limitation (multirecycling). Unlike the vast majority of reactors currently in operation or under construction worldwide, which use only about 1% of natural uranium, fast neutron reactors are able to use more than 80% of the uranium resource. For instance, the current stockpile of depleted uranium available on French territory could feed the French needs for electricity production at the current rate for thousands of years.
- Fast neutron reactors are an intensive energy source, the process of which does not emit greenhouse gases.
- Fast neutron reactors are able to burn minor actinides to produce electricity, thus reducing the quantity, the half-life and the toxicity of the ultimate waste.

The Generation IV International Forum (GIF) selected six concepts as candidates for 4th generation reactors, to be studied in an international collaborative framework:

- Fast neutrons: SFR, Gas-cooled Fast Reactor (GFR), Molten Salt (Fast) Reactor (MS(F)R), Lead-cooled Fast Reactor (LFR);
- Thermal neutrons: Super Critical Water-cooled Reactor (SCWR), Very High Temperature Reactor (VHTR).

The GIF defined an R&D work plan to foster the necessary innovations for the development of such systems. It is worth noting that the technological maturity of the concepts selected by the GIF is very variable.

As far as the CEA is concerned, the efforts are focused on SFRs, and to a lesser extent on GFRs with emphasis on innovation in fuel materials and the safety demonstration over a very long term perspective. CNRS is doing some studies on MSFRs.

Beyond differences of technological maturity, GIF concepts show advantages and drawbacks. In the French case, the focus is on fast neutron technologies, excluding VHTR or SCWR concepts. The MSR concept shows difficulties in the safety demonstration and on the operability side, since there is no classical first barrier for the fuel, putting serious doubts on the industrial operation. As far as the LFR is concerned, corrosion issues and the management of liquid lead at high temperature make this concept less attractive than the SFR, whose feasibility at industrial size was already proven in the past, including the safety demonstration. Based on the accumulated operating experience of more than 400 reactor-years, the SFR shows the best potential to reach 4th generation criteria for industrial deployment by the middle of the 21st century, or even earlier if needed.

The GFR needs the development of a refractory fuel composed of uranium–plutonium carbide fuel pellets with silicon carbide ceramics cladding. The fuel represents the key element of the safety demonstration in case of loss of heat removal systems or in case of depressurization of the primary circuit.

That is why the CEA is working on two types of fast neutron reactor:

— The CEA is contributing to fuel and safety studies for the experimental reactor project ALLEGRO with a thermal power of 80 MW(th), to be built around 2025–2030 in central Europe by a consortium comprising the Czech Republic, Hungary, Poland and Slovakia.
— The CEA is in charge of the 4th generation, SFR ASTRID, with an electrical output power of 600 MW(e), to be put into operation around 2025.

In 2006, in the framework of the law on sustainable management of radioactive materials and waste, a coherent programme towards the development of a 4th generation sodium-cooled fast neutron reactor was set up. A collaborative R&D programme was launched between the CEA, AREVA and EDF, focusing on innovations derived from the past experience of SFRs at national and international levels. At the end of 2009, in a broader investment plan for the future, ASTRID was identified as one of the priorities to receive governmental funding. In 2010, the finance law put into place the multiannual budget for the ASTRID programme and an agreement was signed between CEA and the French Government awarding 650 M€ to the CEA to conduct the ASTRID R&D and design studies, including the development of associated R&D facilities.

The objectives that have been set for the ASTRID programme give high priority to an innovative design, taking advantage of the experience accumulated in former reactors, but showing clear differences with regard to weaknesses that were identified in the past. Moreover, the support by the French Government to the ASTRID programme is associated with the strong requirement for gathering industrial partnerships around the project, so that innovation can also come from the industry, together with the integration of industrial concerns from the beginning of the design.

The ASTRID reactor is seen as an integrated technology prototype designed for industrial-scale demonstration of 4th generation SFR safety and operation. That means that extrapolability of the major technical options and of the safety demonstration is of outmost importance. The reactor will also provide some irradiation capacities especially in order to validate the expected properties for advanced fuel that may be needed for commercial deployment and the ability to burn minor actinides at larger scale than previously done in the Phénix reactor. ASTRID will be coupled to the grid with an electrical power of about 600 MW.

The ASTRID programme also includes the facility to manufacture the fuel for the reactor, of limited capacity from 5–10 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.

The schedule associated to 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 were made in 2010 in order to launch the pre-conceptual and the conceptual design, and to start first discussions with the safety authorities. The first phase of the conceptual design was completed in time at the end of 2012 together with the submission to the French Government of a report on the status of R&D studies on separation and transmutation and on new generation reactors. The second phase of the conceptual design was launched at the beginning of 2013 for a duration of 2 to 3 years. This phase will be followed by the basic and detailed design. The objective is to put the reactor into operation around 2025.

3. HIGHER STANDARDS IN TERMS OF SAFETY AND OPERABILITY

ASTRID will integrate operational feedback of past and current reactors. It is seen as a full Generation IV prototype reactor, with strong improvements on safety and operability. Its safety level will be at least as good as current 3rd generation reactors, with advances on core and sodium related issues, and taking into account the necessary lessons learned from the Fukushima accident. On the availibility side, the reactor will reach a high load factor after a learning period [1, 2].

The new safety requirements put tremendous pressure on R&D and design to meet the expectations, such as the objectives of the 2010 WENRA statement [3, 4]. For instance, in spite of a very high level of prevention, severe accidents have to be postulated and mitigation measures have to be incorporated accordingly into the design. More independence is expected between all levels of defence-in-depth and, following the Fukushima accident, the verification of the absence of cliff-edge effects beyond design is pushed to new limits.

Meeting the ambitious, but necessary, objectives in the field of operability (reliability, availability, maintainability) leads to application of proactive methodologies in this field, without limiting the R&D and design effort to obvious targets such as fuel handling.

From the short experience of ASTRID first phase of conceptual design studies (2010–2012), two remarks can be made:

(1) Higher requirements in safety and operability lead to higher costs that cannot be fully recovered by advances in technology. This puts additional pressure on the next phases of the design to optimize the design and to keep the costs to the minimum.
(2) There is a clear link between the level of safety that can be achieved and the maturity of the technology, i.e. the experience accumulated in R&D, design, construction, operation and decommissioning of past reactors. In the field of fast neutron reactors, this gives a strong advantage to the sodium technology, because strengths and weaknesses are well mastered.

4. DRIVING THE R&D BY THE FEEDBACK OF EXPERIENCE

The R&D performed in France on SFRs is done in close collaboration between the CEA and industrial partners AREVA and EDF.

The R&D programme comprises research in four domains of innovations:

— The development of an attractive and safe core, taking into account the specificities of the fast neutrons and sodium, and also able to transmute minor actinides;
— A better resistance to severe accidents and external hazards;
— The search for an optimized energy conversion system reducing the sodium risks;
— The re-examination of the reactor and component 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 and pool designs;
— Advanced energy conversion systems;
— 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 repear (ISI&R): sensors, inspectability, reparability, robotics.

The reports provided the basis for the selection of possible options for ASTRID. Table 1 shows in a non-exhaustive manner major areas where, deriving from the feedback of experience, R&D was conducted and innovative options were identified.

5. R&D IN SUPPORT TO ASTRID

Since the start of the ASTRID design studies in 2010, R&D has been driven by the needs expressed by the project. This has three main consequences:

(1) The R&D teams have a clear view of the deliverables that are expected and their schedule. A contractual link between R&D and the design team is put in place.
(2) The expression of R&D needs is much broader than before because the approach has changed from a phase of selected highlights in R&D to an exhaustive design phase. When competencies and/or resources are unavailable inside the CEA, collaborations are sought.
(3) Very innovative, cutting-edge concepts with little chance of success are abandoned in favour of innovations that are not necessarily visible at first sight but play an important role in detail. ISI&R is a good example where some major advances are obtained by removing welds in certain locations, aligning areas to be inspected with access holes, etc. Instrumentation and control is another example where great benefit can be obtained just by using state-of-the-art technology as opposed to reactors that were designed decades ago.

TABLE 1. INNOVATIVE OPTIONS FOR ASTRID DERIVING FROM R&D AND FEEDBACK OF EXPERIENCE

Feedback of previous SFR	R&D directions	ASTRID orientations
Core reactivity Issue of sodium void coefficient → Safety	Optimization of core design to improve natural behaviour in case of abnormal transients Exploration of heterogeneous cores	CFV core (patented in 2010): innovative approach, very low or negative overall sodium void coefficient Better natural behaviour of the core, for instance in case of loss of cooling (e.g. due to loss of supply power)
Sodium–water reaction → Safety — availability	Modular steam generators Steam generators with sodium in tubes Gas energy conversion system (nitrogen in place of steam/water)	Limitation of total released energy in case of sodium–water reaction Limitation of wastage propagation Design studies conducted by ALSTOM. No ‘show stopper’
Sodium fire → Safety	Innovation on sodium leak detection systems R&D on sodium aerosols	Improvement of detection (patent on a detection system integrated in the heat insulation) Close containment (limitation of available oxygen, inert gas)
Severe accidents → Safety	Core catcher R&D on corium and sodium–corium interaction	Core catcher. Several locations are under study (in-vessel, ex-vessel or between the two vessels).
Decay heat removal → Safety	Reactor vessel auxiliary cooling system (extrapolability)	Combination of proofed DHR systems, RVACS

TABLE 1. INNOVATIVE OPTIONS FOR ASTRID DERIVING FROM R&D AND FEEDBACK OF EXPERIENCE (cont.)

Feedback of previous SFR	R&D directions	ASTRID orientations
ISI&R	Simplification of primary system design	
→ Safety — availability	ISI&R is taken into account from design on	
	New techniques: acoustic detection, LIBS, CRDS	
	Signal processing	
	TUSHT (ultrasound, high temperature), high temperature fission chambers, optical fibres, flowmeters for subassemblies	
	Remote handling for inspection or repair	
	Under sodium viewing	

However, the organization that has been put into place by the CEA allows R&D teams to propose innovative design options for the project, such as the low sodium void core, CFV, the gas energy conversion system as an alternative to the water–steam route, or some safety systems. In the next design phases, the objective is to keep this possibility to propose new options, while maintaining a sound project framework, including configuration management.

6. R&D COLLABORATION, INDUSTRIAL COLLABORATION AND INTERACTIONS WITH NUCLEAR SAFETY AUTHORITIES

The CEA alone cannot meet the ambitious requirements set for the ASTRID reactor and is relies on several collaborations at different levels:

- On the R&D side, bilateral or multilateral collaborations with dozens of institutes at national, European or broader level. The objectives of the following years will be to emphasize these collaborations and to propose a better integration of this collaborative work into the ASTRID project needs.
- On the design side, the industrial collaboration with 9 main partners is a very valuable asset to the success of ASTRID [1]. Better than in a commercial relationship, the collaborative scheme allows more bottom-up R&D and innovations that are proposed by the industrial partners to the CEA. For instance, strong improvements in the gas energy conversion system were proposed by ALSTOM, feedback of EPR construction will be integrated by BOUYGUES into the civil engineering studies, etc. The objective of the coming years will be to expand this collaborative circle of industrial partners.

The industrial partners are also useful to help the CEA to verify that ASTRID will meet the expectations in terms of operability, since they usually do business with customers to whom reliability, availability and maintainability are essential.

On the side of safety, a very powerful verification of the requirements will be done by the French Nuclear Safety Authority (ASN) and its technical support, IRSN. Discussions started in 2010 and have entered a more formalized phase since the submission by the CEA in June 2012 of the ASTRID safety orientations document. These orientations will be reviewed by the Advisory Committee for Reactors (GPR) in June 2013. The conclusions of the review will be incorporated in the safety options for ASTRID and in the second phase of conceptual design.

7. CONCLUSION

Fast neutron reactors have a large potential as a sustainable energy source. In particular, SFRs with a closed fuel cycle and potential for minor actinide burning allow improved use of natural resources and minimization of high level waste. Among the fast reactor systems, the SFR has the most comprehensive technological basis as result of the experience gained from decades of worldwide operation of several experimental, prototype and commercial size reactors. Currently, in France, a medium size (600 MW(e)) power demonstrator named ASTRID has been proposed and endorsed at the European level as the reference concept for Generation IV fast reactors.

Deriving from the feedback of experience, very high levels of requirements have been set for the ASTRID reactor. Innovations are needed to further enhance safety, reduce capital cost and improve efficiency, reliability and operability, making the Generation IV SFR an attractive option for electricity production. This puts great pressure not only on the R&D and design teams at CEA but also on the CEA's R&D and industrial partners. During the first phase of ASTRID conceptual design (2010–2012), promising innovation options have been identified. They will be further developed in the next phases of ASTRID design studies, keeping a strong interaction between design and R&D.

REFERENCES

[1] LE COZ, P., "The ASTRID Project : Status and future prospects" , CN-199-261, paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[2] LO PINTO, P., "Safety orientations during ASTRID conceptual design phase", CN-199-267, paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[3] WESTERN EUROPEAN NUCLEAR REGULATOR'S ASSOCIATION, Statement on Safety Objectives for New Nuclear Power Plants, November 2010, http://www.wenra.org/media/filer_public/2012/11/05/wenra_statementonsafetyobjectivesfornewnuclearpowerplants_nov2010.pdf

[4] WESTERN EUROPEAN NUCLEAR REGULATOR'S ASSOCIATION, Safety Objectives for New Power Reactors, Study by WENRA Reactor Harmonization Working Group, December 2009.

OVERVIEW OF LEAD BASED REACTOR DESIGN AND R&D STATUS IN CHINA

YICAN WU, YUNQING BAI, YONG SONG, QUNYING HUANG, HONGLI CHEN, GANG SONG, LIQIN HU, JIEQIONG JIANG
Institute of Nuclear Energy Safety Technology,
Chinese Academy of Sciences,
Beijing, China

Abstract

Liquid lead or lead based alloy is a potential candidate coolant for fast reactors and accelerator driven system (ADS) subcritical reactors because of its many unique nuclear, thermophysical and chemical attributes. The Chinese Academy of Sciences (CAS) launched an engineering project to develop ADS system and lead based reactors. A series of China LEAd based Reactors (named CLEAR) design, the lead based experimental loops (KYLIN series Pb–Bi loops and DRAGON series PbLi loops), a high intensified D-T neutron generator (HINEG) and structure material (CLAM) were developed by the Institute of Nuclear Energy Safety Technology. In this paper, the CLEAR design and R&D activities are presented.

1. INTRODUCTION

Nuclear power is playing an important role in China energy safety and environmental protection. However, the nuclear fuel shortage and the spent fuel accumulation issues will restrict the healthy development of nuclear energy in China. So the innovative nuclear system, such as the fast reactor, accelerator driven system (ADS) and fusion reactor is expected to solve these problems.

Liquid lead or lead based alloy (lead based material) is an attractive candidate coolant for fast reactors, ADS subcritical reactor and fusion reactor because of its many unique nuclear, thermophysical and chemical attributes. In addition, lead and bismuth can produce copious spallation neutrons when bombarded with energetic protons. This makes lead based material one of the top candidates for a high power spallation target and coolant in an accelerator driven subcritical system and fast reactors [1, 2]. Since the 1960s, the lead based reactor has been gradually paid more and more attention internationally, and many lead based material cooled reactor concept designs and lead based coolant technology R&D have been carried out in some countries [3–6].

The Chinese Academy of Sciences (CAS) launched an engineering project to develop the ADS system and lead based reactors named China LEAd based Reactor (CLEAR) series. In this programme, CAS plans to develop the lead based reactors through 3 phases which are a 10 MW(th) lead–bismuth cooled research reactor (CLEAR-I) to be built in 2010s, a 100 MW(th) lead based experimental reactor (CLEAR-II) to be built in 2020s, and a 1000 MW(th) lead based demonstration reactor (CLEAR-III) to be built in 2030s. As a pre-testing facility, a lead based zero power reactor (CLEAR-0) is required to be built before CLEAR-I construction and operation.

However, lead based materials as a coolant for a reactor have several challenges which are needed to be considered and verified. The emerging worldwide interests in the applications of lead based materials led to much R&D activity in the fields of materials compatibility, neutronics, thermohydraulics, physical chemistry, and safety characteristics, etc. KYLIN series lead–bismuth loops, a high intensified D-T neutron generator (HINEG) and structural material (CLAM) were developed by the Institute of Nuclear Safety Technology, CAS (INEST・FDS Team, CAS).

Lead based coolant was used not only for the fast reactor and ADS system, lead–lithium eutetic also has advantages when used in a fusion reactor blanket. The FDS series lead–lithium cooled fusion reactor blanket concept design [7] and the DRAGON [8] series lead–lithium experiment loops were performed by the FDS team in China.

2. CLEAR SERIES LEAD BASED REACTOR DESIGN

2.1. China LEAd based research Reactor (CLEAR-I)

CLEAR-I was designed to be operated at two modes: critical operation for fast reactor technology tests and accelerator driven subcritical operation for the ADS system. The critical operation mode reactor was named CLEAR-IA, and the subcritical operation mode reactor was named CLEAR-IB driven by spallation neutron created by proton accelerator. The research object of CLEAR-IA is to validate the neutronics, thermohydraulics and safety characteristics of the lead–bismuth cooled reactor, and test the fuel and material technologies. The research object of CLEAR-IB is to test the integration technology of the ADS system. The isotope neutron source can be replaced by a spallation target to change CLEAR-IA to CLEAR-IB. Table 1 shows the main design parameters of CLEAR-I.

In the reference parameters design of CLEAR-I, pool-type configuration is selected, where the reactor vessel is placed inside a safety vessel. The thermal power is 10 MW and no electric power was generated. Lead–bismuth eutectic

(LBE) is adopted as the primary coolant and UO_2 with a ^{235}U enrichment of 19.75% is chosen as fuel. Hexagonal wrapped fuel assemblies are used in a hexagonal lattice core; the cladding material is SS316Ti and structural material is SS316L. Considering the characteristic of low power density of reactor and large expansion coefficient of lead–bismuth, the primary cooling system is designed to rely on natural circulation entirely, which improves the safety characteristics of the reactor. Large diameter pins are used to achieve higher fuel volume fraction and lower core pressure drop. Negative fuel temperature coefficient and negative coolant reactivity feedback are achieved through neutronics design. Two independent secondary cooling systems with liquid water as coolant are designed. Air is used as the final heat sink by water/air heat exchanger. Decay heat is removed by passive reactor vessel air cooling system. Figure 1 shows the structure design for the CLEAR-I reactor.

TABLE 1. THE MAIN PARAMETERS OF CLEAR-I REFERENCE SCENARIO

Thermal power (MW)	10
Active height (m)	0.8
Active diameter (m)	1.05
Fuel (235U enrichment)	UO_2 (19.75%)
Primary coolant	LBE
Inlet/outlet temperature (°C)	260/390
Coolant drive type	Natural circulation
Second coolant	Water
Heat sink	Air cooler
Cladding material	316 Ti
Structure material	316L

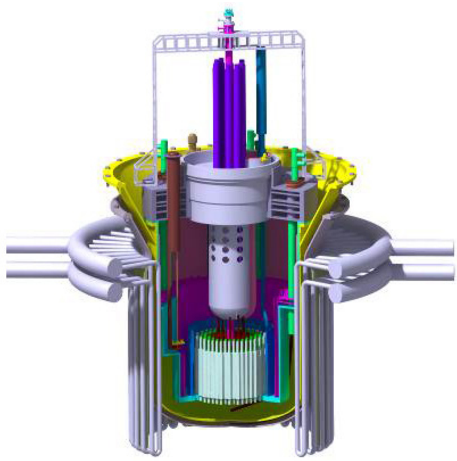


FIG. 1. Overview of CLEAR-I structural design.

2.2. China LEAd based experiment Reactor (CLEAR-II)

For the second stage of the CAS ADS programme, an ADS experimental facility will be built to test the high power density ADS system and used as a fast neutron irradiation facility. A 100 MW(th) lead or lead–bismuth cooled experimental reactor named CLEAR-II will be built coupled with a proton accelerator of ~600–1000MeV/~10mA and a lead–bismuth spallation target. Based on CLEAR-II, the high power ADS system design, construction and operation control technology will be enhanced. To improve the reactor core neutron flux and power density, the nuclear fuel will use high enrichment MOX fuel. Fuel assemblies can partially be replaced by minor actinide assemblies to research the nuclear waste transmutation mechanism. CLEAR-II can also be used as a high power spallation neutron source or high neutron flux reactor, and carry out material irradiation experimental studies. Figure 2 shows the CLEAR-II structural design.

2.3. China LEAd based demonstration Reactor (CLEAR-III)

For the third stage of the CAS ADS programme, CLEAR-III is a lead based demonstration reactor, which aims to demonstrate the technologies of highly efficient nuclear waste transmutation of commercial ADS. For the CLEAR-III reference scenario, an accelerator driven lead based subcritical

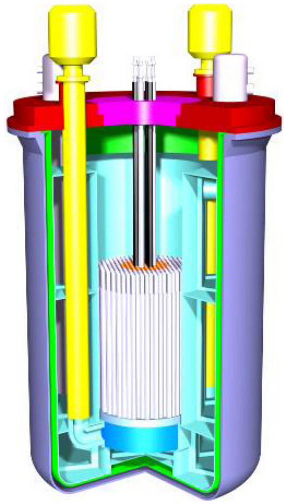


FIG. 2. Overview of CLEAR-II structural design.

reactor for transmutation of long lived high level nuclear waste is developed based on the neutronics, thermohydraulics, and mechanics analysis. The lead and lead–bismuth are also considered as the potential coolants for CLEAR-III to investigate high efficiency power generation and waste transmutation. A linac accelerator produces a proton beam of 1.5 GeV with about 10mA and the proton impinges on the windowless lead–bismuth target in the core central region. The CLEAR-III system is rated at 1000 MW(th) thermal power. Currently, one of the fuel types considered for CLEAR-III is the TRU-Zr dispersion fuel, where TRU-Zr particles are dispersed in a Zr matrix. The advanced ferritic/martensitic steel is selected as the target guide pipe and fuel cladding material owing to its good performance under a high corrosive and radiation environment. Figure 3 shows the illustration of CLEAR-III structure design.

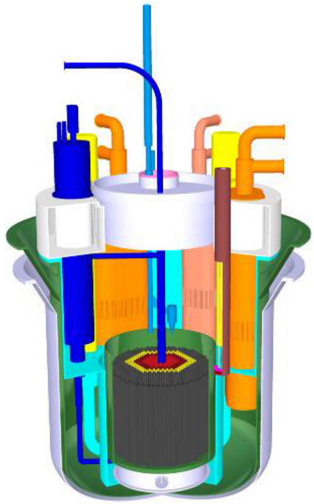


FIG. 3. Overview of CLEAR-III structural design.

2.4. China LEAd based zero power Reactor (CLEAR-0)

In order to validate the design approach, calculating programme, database and support the design and safety analysis for CLEAR series reactors, it is necessary to carry out a series of zero power neutronics experiments. Therefore, the China LEAd based zero power Reactor (CLEAR-0), a zero power fast spectrum experimental facility, was designed to meet this requirement. Figure 4 shows the structural design for the CLEAR-0. Under this conceptual design, the main structure of this experimental facility sits in a reactor pit, above which there is a removable biological shield. The cores studied in CLEAR-0 comprise a lattice of standard assemblies. By changing the simulation materials loaded in standard assemblies, CLEAR-0 can simulate various cores. Meanwhile, it is worth noting that two reactor trip systems based on different mechanisms ensure CLEAR-0 safety. The experiments on CLEAR-0 can be divided into two phases. The initial phase will focus on critical experiments. Then, the facility will be coupled with a neutron generator to perform external source driven subcritical experiments.

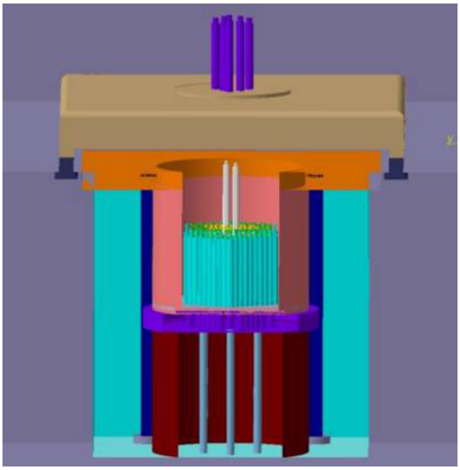


FIG. 4. Overview of CLEAR-0 structural design.

2.5. New conceptual lead based reactor

CLEAR series reactors can be used not only for ADS and fast reactors, but also for innovative application. Series lead based reactors' new conceptual design is being developed by the FDS team, including CLEAR-T for fusion reactor tritium production, CLEAR-SFB for spent fuel burning, CLEAR-Th for thorium fuel recycle, CLEAR-H for hydrogen production and CLEAR-SR for a small modular reactor.

3. R&D ACTIVITIES OF KEY TECHNOLOGIES

3.1. KYLIN series lead–bismuth loops

Even though lead–bismuth has the advantages of excellent neutron properties, including good thermal conductivity and low chemical activity, there are also some key issues to be experimentally investigated, such as compatibility of LBE with materials, the flow and heat transfer characteristics of LBE, chemical control, instrumentation technology, etc. The liquid lead–bismuth loops are the necessary facilities for the material technologies and thermohydraulic challenges. The KYLIN series lead–bismuth experimental loops and verification facilities for the CLEAR reactor are being designed and built in the INEST. The main parameters of KYLIN loops are listed in Table 2.

TABLE 2. KYLIN SERIES LBE EXPERIMENTAL LOOPS FUNCTION AND DEVELOPMENT PLAN

Loop name	**Type***	**Function**	**Temperature**	**Time**
KYLIN-I	TC	Compatibility test under flowing Pb–Bi	480–550ºC	2010
KYLIN-II	FC/TC	Materials test, thermohydraulics experiment, safety experiment	480~800ºC	2010–2013
KYLIN-III	FC	Thermohydraulics verification facility for CLEAR-I	200–550ºC	2012–2014
KYLIN-ST	ST	Compatibility test in the static Pb–Bi	200~800ºC	2010
KYLIN-RT	RT	Compatibility test in the rotation flowing Pb–Bi	480~600ºC	2010

* TC: thermal convection, FC: forced convection, ST: static test, RT: rotation test.

KYLIN-I (see Fig. 5), KYLIN-ST and KYLIN-RT are compatibility test platforms for the study of the structural materials exposed with LBE. The KYLIN-I is a kind of thermal convection loop and has been operated for more than 5000 h to test the corrosion behaviour of candidate materials, such as SS316L and CLAM steel, in LBE [9]. The KYLIN-ST and KYLIN-RT is built to test a series of candidate materials, both in static and flowing lead–bismuth under different velocities.

KYLIN-II (see Fig. 6) is a large multifunctional lead–bismuth experiment loop, which has three independent functional zones for material tests, thermohydraulic tests and safety experiments for lead–bismuth with pressurized water. The maximum temperature in the test section of the KYLIN-II materials loop is 800°C for testing the structural materials in some accident conditions, and the temperature 800°C can be achieved by flowing channel inserts, such as SiC composites and other high temperature alloys, the maximum velocity in test section is 6 m/s for testing the impeller of a mechanical pump. The gas phase oxygen control will be applied to this facility to enhance the anti-corrosion of the structural materials.

The KYLIN-II thermohydraulic experimental loop has natural convection and forced convection operation modes to study lead–bismuth flow and heat transfer characteristics and validate the CLEAR-I thermohydraulic design, key components test, and fuel bundle tests on a scale of 1:1 for CLEAR-I.

The KYLIN-II safety facility could be used to simulate a steam generator tube rupture (SGTR) event for CLEAR series reactors to investigate the sub-phenomena of SGTR and support the qualification of license for a CLEAR-I heat exchanger tube rupture event. Meanwhile, the results from these simulations could also help to develop mitigation and innovative device against a SGTR event and be used for verification and validation of codes.

KYLIN-III is an integrated thermohydraulic verification facility for CLEAR-I pool type natural convection and flow distribution. This facility will be built in 2014.

3.2. Highly Intensified D-T NEutron Generator (HINEG)

Besides the materials compatibility and thermohydraulics experimental study, it is also important to have a lead based reactor neutronics experiment study. Highly intensified neutron generators are very useful tools. Construction of HINEG for neutron experiments and software validation will be finished by INEST in about 2014. It aimed to perform reactor neutronics design verification, materials radiation experiments, radiation protection and environmental impact study and nuclear technology application study. The maximum D-T neutron intensity will be 3×10^{13}n/s. The lead based material and fuel assembly will be combined with HINEG to test lead based reactor neutronics characteristics as well as to validate software and nuclear databases.

FIG. 5. KYLIN-I Pb–Bi loop.

a) Materials loop b) Thermohydraulic loop c) Safety loop

FIG. 6. KYLIN-II Pb–Bi loop.

3.3. Structural materials

Reduced activation ferritic/martensitic steels with good irradiation swelling resistance and thermophysical and thermomechanical properties have been considered as the important candidate structural materials for a lead based material cooled reactor. China low activation martensitic (CLAM) [10] steel is being developed by the INEST • FDS team, the recent R&D activities on CLAM steel mainly include production of large scale heat, properties testing and fabrication of small mockups. To promote its industrial application, larger ingot smelting work (i.e. a heat of 4.5 t ingot) was carried out (Fig. 7). Some key issues, such as composition segregation, have been investigated. To evaluate the properties of the industrialized production scale steel, the compositions measurement, property tests and microstructure analysis were performed. High energy spallation neutron irradiation experimental doses up to 20 dpa on CLAM have been carried out.

4. SUMMARY

Lead based materials have excellent heat transfer ability and safety features and form important candidate materials for advanced nuclear systems. The CAS launched the Strategic Priority Research Programme on the Future Advanced Nuclear Fission Energy-ADS Transmutation System to develop lead based reactors (2011), and plans to build the first lead–bismuth cooled reactor in China before 2020. The conceptual designs of the CLEAR series have been performed. The engineering design of the natural circulation lead based research reactor CLEAR-I and lead based zero power reactor are under way. Natural circulation lead–bismuth experimental loops KYLIN-I/RT/ST for material corrosion experiment and multi-functional large scale lead–bismuth experimental loops KYLIN-II will be constructed soon. Construction of the HINEG for neutron experiments and software validation will be finished by INEST in about 2014. CLAM steel is being developed and tests in the lead–bismuth and neutron irradiation environments are in progress.

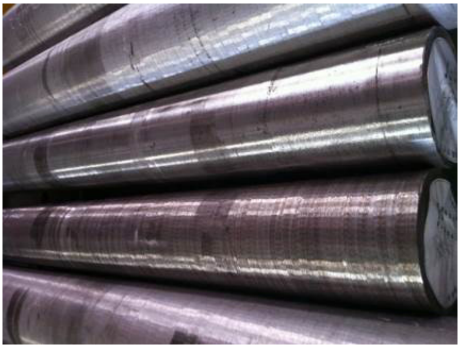


FIG. 7. CLAM 4.5 t ingot.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Accelerator driven systems: Energy generation and transmutation of nuclear waste. Status report, IAEA-TECDOC-985, IAEA, Vienna (1997).

[2] NING L., Lead-alloy coolant technology and materials e technology readiness level evaluation, J. Prog. Nucl. Energy **50** (2008) 140–151.

[3] ABDERRAHIM, H.A., et al., European Experimental ADS for R&D Applications status at Mid-2005 and prospective towards implementation, J. Nucl. Sci. Technol., **44** (2007) 491–498.

[4] UNITED STATES DEPARTMENT OF ENERGY, A Roadmap for Developing Accelerator Transmutation Accelerator Transmutation of Waste (ATW) Technology, A Rep. to Congress, Rep. DOE/RW-0519, USDOE, Washington, DC (1999).

[5] KURATA, Y., et al., The accelerator driven system strategy in Japan, J. Nucl. Mater. **301** (2002) 1–7.

[6] PARK, W.S., et.al., HYPER (Hybrid Power Extraction Reactor): A system for clean nuclear energy, J. Nucl. Eng. Des. **199** (2000) 155–165.

[7] WU, Y., et al., Conceptual design activities of FDS series fusion power plants in China，J. Fus. Eng. Des. **81** (2006) 2713–2718.

[8] HUANG, Q., et al., Progress in compatibility experiments on lithium-lead with candidate structural materials for fusion in China, J. Fus. Eng. Des., **84** (2009) 242–246.

[9] WU, Y., et al., Preliminary experimental study on the corrosion of structural steel in liquid lead Bismuth .oop, Chin. J. Nucl. Sci. Eng. **30** (2010) 238–243.

[10] QUNYING, H., Progress in development of CLAM steel and fabrication of small TBM in China, J. Nucl. Mater. **417** (2011) 85–88.

FAST REACTOR TECHNOLOGIES, COMPONENTS AND INSTRUMENTATION

Chairpersons

J.-M. HAMY
France

P. CHELLAPANDI
India

C. GRANDY
United States of America

THE LEAD FAST REACTOR: DEMONSTRATOR (ALFRED) AND ELFR DESIGN

M. FROGHERI, A. ALEMBERTI, L. MANSANI
Ansaldo Nucleare,
Genoa, Italy

Abstract

The LEADER project has the objective of designing a commercially viable reactor, for large scale electricity production (ELFR), based on lead coolant technology. The industrial deployment of ELFR requires a scaled demonstrator reactor, with the objectives of demonstrating the achievement of the required safety standards, of assessing the economic competitiveness of lead technology and of validating the engineering options and materials selection. The paper summarizes the project and discusses the main design features of the demonstrator ALFRED (Advanced Lead Fast Reactor European Demonstrator) and of the ELFR plant.

1. INTRODUCTION

The lead cooled fast reactor (LFR) technology has a great potential to fulfil all the main goals established by the Generation IV International Forum (GIF) for the next generation of nuclear power plants [1]. The LFR is based on a closed fuel cycle for efficient conversion of fertile uranium and management of actinides (enhanced sustainability), the inert nature of the coolant provides important design simplification (improved economics) and allows for designing decay heat removal systems based on well known light water technology and passive features (increased safety). Moreover, MOX, reference fuel for ELFR and ALFRED, constitutes a very unattractive route for diversion or theft of weapons-usable materials and provides increased physical protection against acts of terrorism (non-proliferation and physical protection).

At the international level, GIF identified in its technology roadmap the six most promising GEN IV systems, as well as the research and development (R&D) activities needed to establish their feasibility and performance capabilities. In particular, with regard to lead cooled reactors, GIF considered several options: a small transportable system of 10–100 MW(e) size (SSTAR — USA) that features a very long core life, a system of intermediate size (BREST 300 — Russian Federation), and a larger system rated at about 600 MW(e) (ELFR — EU) and intended for central station power generation.

In Europe, the European Commission organized the Sustainable Nuclear Energy Technology Platform (SNETP) that, through its Strategic Research Agenda, promoted the development of fast reactors with closed fuel cycle. The roadmap proposed by the European Sustainable Nuclear Industrial Initiative (ESNII) includes the lead cooled fast reactor as an alternative technology in the short term to be developed in parallel with the sodium cooled fast reactor while the gas cooled fast reactor is still considered for a long term option.

A major step in the development of a Lead Cooled Critical Fast Reactor in Europe started in 2006, when EURATOM decided to fund ELSY (European Lead cooled SYstem). The ELSY project [2], coordinated by Ansaldo Nucleare, developed a very innovative pre-conceptual design of an industrial plant for electricity production able to close the fuel cycle.

The LFR development continued with the LEADER project [3] (Lead cooled European Advanced DEmonstration Reactor), started on April 2010 in the framework of the EU 7th Framework Programme. It takes into account the indications emerged from SNETP, as well as the main goals of ESNII, and aims for the development to a conceptual level of an LFR industrial size plant and of a scaled demonstrator of the LFR technology — ALFRED (Advanced Lead Fast Reactor European Demonstrator). The project involves 17 partners from Industry, research organizations and universities. The total effort is 502 person-months over a period of 36 months.

The open issues and the safety concerns, emerged during the previous projects, have been analysed in the LEADER project and a new set of design options and safety provisions proposed, in order to define the reference ELFR industrial plant configuration. The updated reactor configuration is then used to design a low cost and fully representative scaled down demonstrator of a suitable size.

ALFRED, in the role of LFR demonstrator, will show the viability of the LFR technology for use in a future commercial power plant, being the first link of the technology chain connected to the electrical grid. It is expected to significantly reduce uncertainties in construction and licensing.

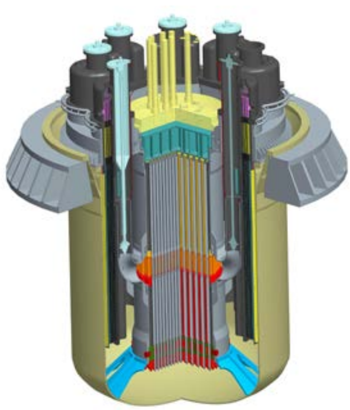
The objectives regarding neutronics are mainly related to the need for investigating the feasibility of a core with closed fuel cycle (adiabatic core), which is a very crucial issue for an actual sustainability with respect to both natural resources and environmental impact. Moreover, the demonstrator will confirm that the newly developed and adopted materials, both structural material and innovative fuel, are able to sustain high fast neutron fluxes and high temperatures.

The demonstration of safe and reliable operation is a key goal, thus the reactor core and primary coolant path should be as prototypical as possible, as well as the secondary circuit and the balance of plant (BOP). On the other hand, the aim of proceeding in the near future to a detailed design followed by the construction phase requires the use of components/technologies already available in the short term. As a consequence, ALFRED is designed to be as close as possible to the reference industrial size plant, but, where needed, proven and already available solutions are adopted, although different from the design proposed for ELFR.

2. ALFRED REFERENCE CONFIGURATION

The configuration of the primary system is pool-type. This concept permits containment of all the primary coolant within the reactor vessel, thus eliminating all problems (e.g. no LOCA initiated by external pipes breaks, no need of intermediate loop reducing cost and earthquake issue) related to out-of vessel circulation of the primary coolant (Fig. 1).

The reactor assembly presents a simple flow path of the primary coolant, with a riser and a downcomer. The heat source (the core), located below the riser, and the heat sink (the steam generators (SG)) at the top of the downcomer, allow an efficient natural circulation of the coolant. The primary coolant moves upward through the pump impeller to the vertical shaft, then enters the SG through the lead inlet holes, flows downwards on the shell and exits the steam generator.



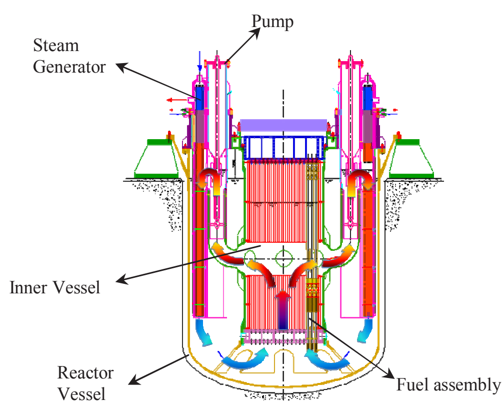


FIG. 1. ALFRED 3-D sketch and reactor block vertical sections.

2.1. Primary system arrangement

The reactor vessel is cylindrical with a torospherical bottom head. It is anchored to the reactor cavity from the top, by means of a vessel support. The upper part is divided in two branches by a “Y” junction: the conical skirt that supports the whole weight, and the cylindrical one that supports the reactor cover. A cone frustum, welded to the bottom head, has the function of bottom radial restraint of inner vessel.

A steel layer covering the reactor pit, constitutes the safety vessel. The dimensions of the gap between the safety vessel and the reactor vessel are sufficient for the in-service inspection tools. The safety vessel is cooled by the same system that cools the concrete of cavity walls. This system is inserted inside the concrete and is independent of the reactor cooling systems. This design solution mitigates the consequences of through-wall cracks with leakage of lead: any reactor vessel leakage is discharged into the safety vessel. The reactor vessel and the safety vessel are arranged in such a manner that in case of a reactor vessel leak, the resulting primary coolant always covers the SG inlet and the lead flow path is maintained.

The inner vessel (Fig. 2) has two main functions: fuel assembly (FA) support and separation between hot plenum and cold plenum. It is fixed to the cover by bolts and is radially restrained at the bottom. Lead flow is guided from the FAs’ outlet towards the primary pump inlet pipes by a toroidal half-ring. Moreover, the pipes that connect the hot zone with the inlet of the primary pump are integrated in the inner vessel. The cylindrical inner vessel has a double wall shell: the outer thick wall has a structural function, while the inner thin wall follows the core section profile. The core lower grid is a box structure with two horizontal perforated plates connected by vertical plates. The plate holes are the housing of FAs foots and the plates distances must be sufficient to assure the verticality of FAs. The diagrid is mechanically connected to the inner vessel with pins (possible removal/replacement during reactor lifetime). The core upper grid is a box structure like the lower grid but stiffer. It has the function to push down the FAs during the reactor operation. A series of preloaded disk springs press each FA on its lower housing. A hole is present for each disk to allow the passage of instrumentation (i.e. thermocouples).

2.2. Core

The adopted core configuration is constituted by hexagonal FAs with T91 wrapper. It utilizes MOX as fuel with hollow pellets and in order to improve the natural circulation a low active height is used. The cladding and the spacer grids are in 15-15/Ti. The total power is 300 MW(th).

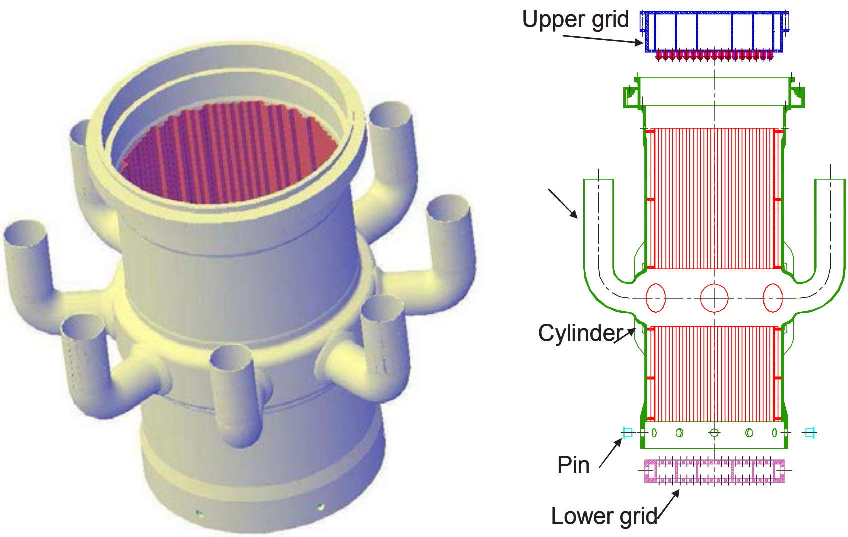


FIG. 2. Inner vessel: 3-D sketch and axial section with FAs inside.

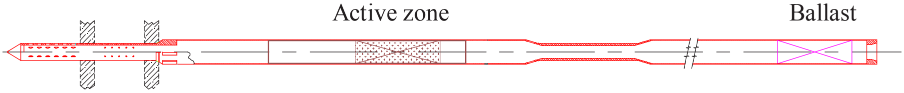


FIG. 3. FA sketch.

The core scheme is made of 171 FAs, 12 control rods and 4 safety rods, surrounded by 108 dummy elements (ZrO_2-Y_2O_3) shielding the inner vessel.

Each FA (Fig. 3) is about 8 m long and consists of 127 fuel pins, fixed to the bottom of the wrapper and restrained sideways by grids. Tungsten deadweight (Ballast) prevents buoyancy forces in lead. Upper elastic elements (cup springs) prevent lifting induced by hydrodynamic loads and accommodate axial thermal expansions. The FAs upper end extends beyond the lead free surface in the cover gas for easy inspection and handling. In this way it is possible to make the refuelling with remote handling but without the need for in-vessel refuelling machines.

ALFRED is equipped with two diverse, redundant and separate shutdown systems (adapted from the one that is under investigation in the frame of the CDT-MYRRHA project [4]):

(1) The control rod system is used for both normal control of the reactor (startup, reactivity control during the fuel cycle and shutdown) and for SCRAM in the case of emergency. The control rods are extracted downward and rise up by buoyancy in the case of SCRAM. The control mechanism pushes the assembly down with a ball screw, placed, with its motor and resolver atop the cover (at cold temperature (<70°C)), and protected from radiation by a shielding block. The actuator is coupled to a long rod by an electromagnet. When the coupling electromagnet is switched off (in case of a SCRAM), the absorber assembly and the rod are free to rise up. Control rods use a 19 pin absorber bundle, cooled by the primary coolant flow. These pins are fitted with a gas plenum collecting the helium and tritium, produced by nuclear reaction of B-10

(2) The safety rod system is the redundant and diversified complement to the control rods for SCRAM only. The absorber bundle stays in the primary coolant. The rod is extracted upwards and inserted downwards against the buoyancy force. The absorber is inserted by the actuation of a pneumatic system. In the case of loss of this system, a tungsten ballast will force the absorber down by gravity in a slow insertion.

For both systems, the materials considered are B_4C enriched in B-10 at 90% as absorber, T91 for the guide tube, 15-15/Ti for the clad and ZrO_2 (95%)–Y_2O_3 (5%) for the insulator and the reflector.

2.3. SG and primary pump unit

The steam generator and primary pump are integrated into a single vertical unit. Eight SG/primary pump units are located in the annular space between the cylindrical inner vessel and the reactor vessel wall. The primary pump is placed in the hot side of the steam generator, having its mechanical suction in the hot pool inside the inner vessel. The primary coolant moves upwards through the pump impeller to the vertical shaft, then enters the SG through the lead inlet holes, flows downwards on the shell and exits the steam generator. The pump motor is located above the reactor roof.

Each SG consists of a bundle of 542 bayonet tubes immersed in the lead vessel pool for six metres of their length. The bayonet tube is a vertical tube with external safety tube and internal insulating layer, composed of 4 concentric tubes (Fig. 4): slave tube, inner tube, outer tube and outermost tube. The internal

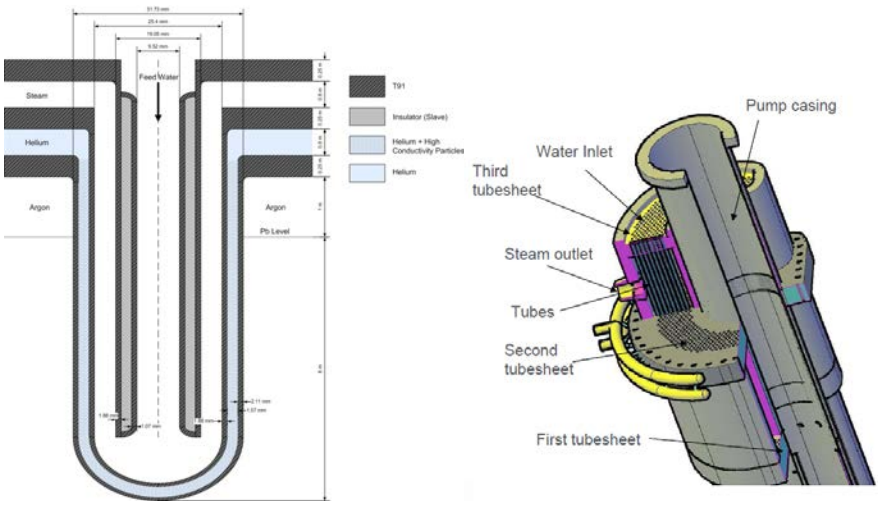


FIG. 4. bayonet tube configuration and SG 3-D scheme.

insulating layer (delimited by the slave tube) has been introduced to ensure the production of superheated dry steam: in fact, without an insulating system, the high ΔT (≈115°C) between the rising steam and the descending feedwater would promote steam condensation in the upper part of the steam generator. The gap between the outermost and the outer bayonet tube is filled with pressurized helium and high thermal conductivity particles, such as synthetic diamonds, to enhance the heat exchange capability. In the case of an external tube break, this arrangement guarantees that primary lead does not interact with the secondary water. Moreover, a tube break can be easily detected by monitoring the helium gap pressure.

The primary pump is located in the hot side of an SG. It is surrounded by the SG tube bundle and its housing is the SG casing. The pump is fixed to the top of SG casing by a bolted joint. This allows easy removal of the component.

Primary pump studies are in progress. Based on analyses performed during previous LFR projects (EUROTRANS, ELSY) an axial pump has been adopted. The primary pump impeller material is still an issue and test campaigns must be performed to select the proper one. Stainless steel cannot be used because the high speed achieved at the top of the impeller blades induces very rapid material corrosion. Maxthal ceramic material has been proposed, but its reliability must be still demonstrated. An alternative solution could be a stainless steel impeller with a ceramic coating.

2.4. Decay heat removal system

The decay heat removal system (DHR) consists of two passive, redundant and independent systems, DHR1 and DHR 2, both composed of four isolation condenser (IC) systems connected to four SGs secondary side (i.e. one IC for each SG). The system design considers the single failure criteria, since three out of four ICs are sufficient to remove the decay heat power.

The DHRs are dedicated safety systems, not used for normal operation. The separation is achieved through placing the two DHRs in physically different locations. A physical structural barrier or another means of protection will be placed between adjacent IC to ensure that failure of one of them could not harm the other one. The diversity requirement has been relaxed (in any case the two DHR systems will be fabricated by diverse manufacturers) due to the high redundancy and considering that the SG tubes bayonet concept allows a continuous monitoring of the SG status.

Both systems are passive (passive execution/active actuation as noted in IAEA-TECDOC-626), with an active actuation through valves equipped with redundant and diverse energy sources (batteries or locally stored energy). Each DHR system must be ready to operate after the reactor trip in order to remove the decay heat power, in case of unavailability of the normal path (i.e. the by-pass to the condenser). The actuation logic will guarantee the actuation of DHR1 first, whereas the DHR2 will actuate only in case of failure of the first system. Moreover, the total number of ICs called on to operate will never overcome the four units, in order to avoid an excessive cooling of the primary coolant leading to fluid solidification. In the case of station black-out, the logic is powered by batteries. However, to completely resolve this issue, a promising improvement to the DHR system is under study to make it completely passive.

Each of the four independent IC subsystems (Fig. 5) consists of:

— One heat exchanger (isolation condenser), constituted by a vertical tube bundle with an upper and lower horizontal header;
— One water pool, where the isolation condenser is immersed, the amount of water contained in the pool is sufficient to guarantee 3 days of decay heat removal operation;
— One condensate isolation valve (to meet the single failure criteria this function will be performed at least by two parallel valves).

Each isolation condenser is connected to an SG, the upper header of the IC is connected to the main steam line and the lower header of the IC is connected to the main feedwater line.

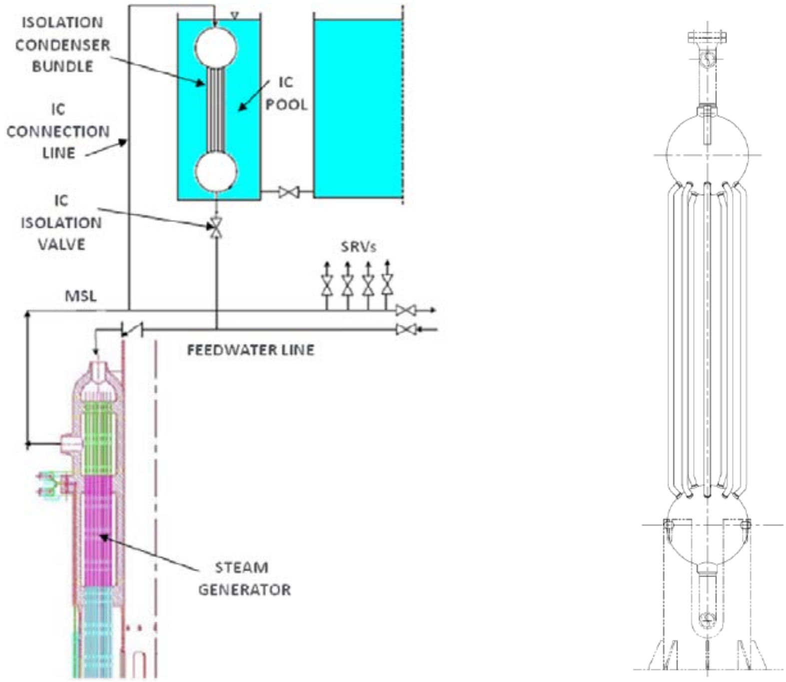


FIG. 5. ALFRED IC scheme and IC bundle.

In normal operation. the isolation valve below the condenser is closed, the condenser is full of water and no heat exchange takes place. As the IC subsystem is called to operate, the feedwater line and steam line are isolated and the condensate isolation valve opens. The subcooled water stored into the IC tube bundle drains into the SG, due to the hydrostatic head existing between the reactor and the IC. The IC tubes and headers become empty and their internal cold surface starts to condense the steam coming from the SG and hence to transfer heat to the cold pool water. The steam condensation causes a pressure reduction which draws other steam from the SG. The water injected into the secondary system from the IC during the draining phase, vaporizes into the SG tube bundle contributing to the secondary system pressure rise: the safety relief valves continue to operate to reject to the atmosphere the excess of steam and to guarantee a secondary side pressure of 195 bar(a). When the IC reaches its steady state condition and starts to remove the thermal power from the primary coolant, the secondary side pressure rise decreases and finally stops leading to the closure of the safety relief valves on the main steam line.

2.5. Secondary system

The secondary system proposed for ALFRED is based on a dual turbine configuration with three extractions in the HP turbine and three more in the LP turbine, with an axial outlet. There will be a reheating with steam from the first extraction and six preheaters supplied with steam from each turbine extraction, as well as a final heater supplied with main steam. This main steam will be adequately throttled so that the feedwater temperature at the inlet of the SG (FWTC heater — feedwater temperature control) can be controlled. In addition, the de-aerator can be fed from the outlet of the HP turbine. The typical redundancy for the condensate and feedwater pumps (2 × 100% pumps) has also been considered.

An auxiliary lead heating system is added. This system would work when the power cycle is not in operation, in order to ensure the minimum temperature of the lead by transmitting heat from the secondary system if it is needed.

A 100% turbine bypass system is included, permitting direct transfer from the reactor to the condenser (bypass mode).

A sketch of the current ALFRED layout with forced draft cooling towers is shown in Fig. 6.

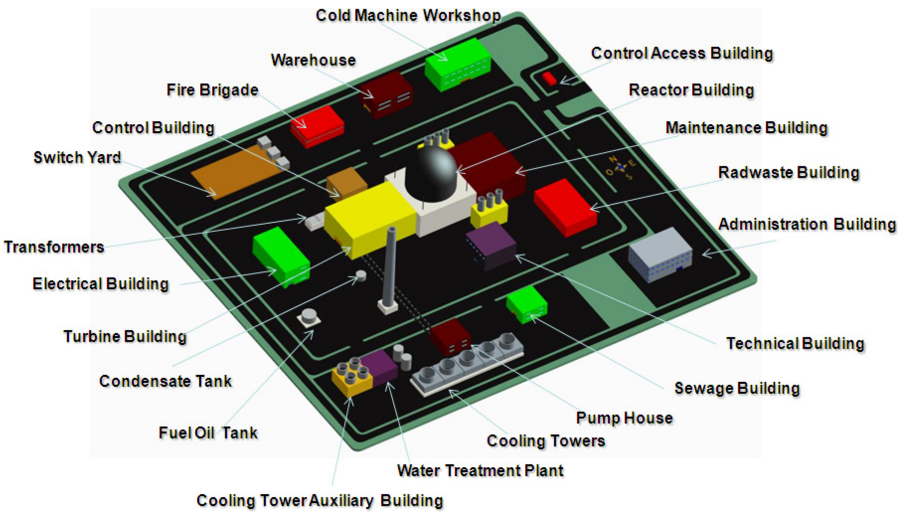


FIG. 6. ALFRED: general layout.

3. ELFR REFERENCE CONFIGURATION

With respect to ALFRED, the ELFR is a larger system (600 MW(e)), since it is intended primarily for central station electricity generation. A sketch of the ELFR reactor block is shown in Fig. 7.

The design of most of its primary system components, such as reactor vessel, FA, inner vessel and core supports, is the same as already described in Section 2 for ALFRED, but with larger dimensions. Some other components are different, since, in order to simplify the design and licensing phases for an early construction of ALFRED, proven and qualified design solutions have been adopted for the demonstrator. In particular:

— SG: In the current ELFR configuration, a once through SG with spiral tubes bundle is considered, whereas ALFRED SG is constituted by double wall straight bayonet tubes, allowing an easy coating and/or surface treatment to speed-up the construction.
— DHRs: In ELFR, the DHRs consists of two diverse systems — the IC and the DHR2 system, constituted of dip coolers immersed in the reactor pool. In ALFRED, considering that the bayonet concept allows a continuous monitoring of the SG status, the diversity requirement has been relaxed. The two DHRs are both constituted by ICs.

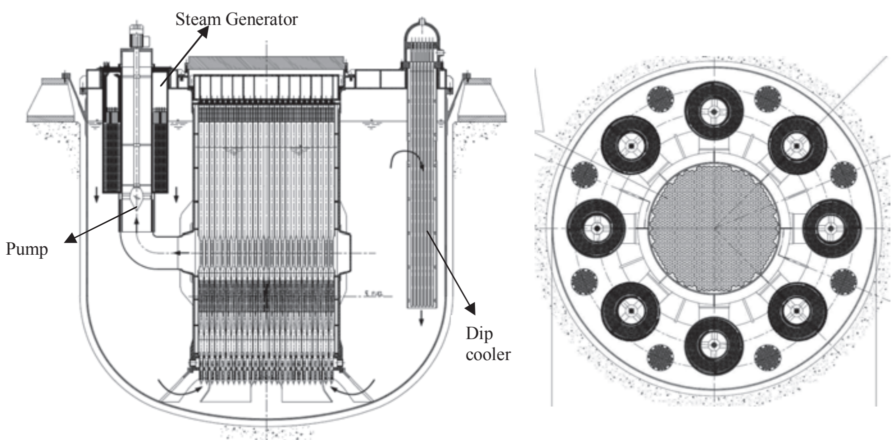


FIG. 7. ELFR reactor block vertical section and plan views.

3.1. ELFR SG

The SG and primary pump are integrated into a single unit and placed vertically through circular penetrations in the reactor roof in the downcomer. The primary pump is placed in the centre of the flat-spiral type SG, having its mechanical suction in the hot leg. The primary coolant moves upward through the pump impeller to the vertical shaft and then transversally (radially) through the SG tubes on the shell side out of the SG to the downcomer through the perforated double-wall casing (Fig. 8).

The SGs' inner and outer shells are composed by two walls connected by welded pins. Both main and companion walls have holes to permit the lead to flow across the tube bundle; the double wall system has been introduced to avoid the propagation of tube rupture effects in the core. The hole positions on the two coupled walls are made so that lead passage is avoided when the walls come in contact. Tests to prove these features are ongoing.

The tube bundle is made of 218 tubes having an average length of 55 m, an OD of 22.22 mm, and a thickness of 2.5 mm. The tubes have a pitch of 24 mm in both the radial and axial directions and are located in staggered planes.

At the ends of each tube, before the connection with the collectors, are located a “Venturi” nozzle flow limiter on the feedwater inlet side and a check valve on the steam outlet side. The “Venturi” nozzle has two functions: it acts as a ferrule during the normal operations and limits the leaking flowrate of the tube to the critical flowrate in case of tube rupture. The check valve on the steam side has the function of avoiding the reverse steam flow.

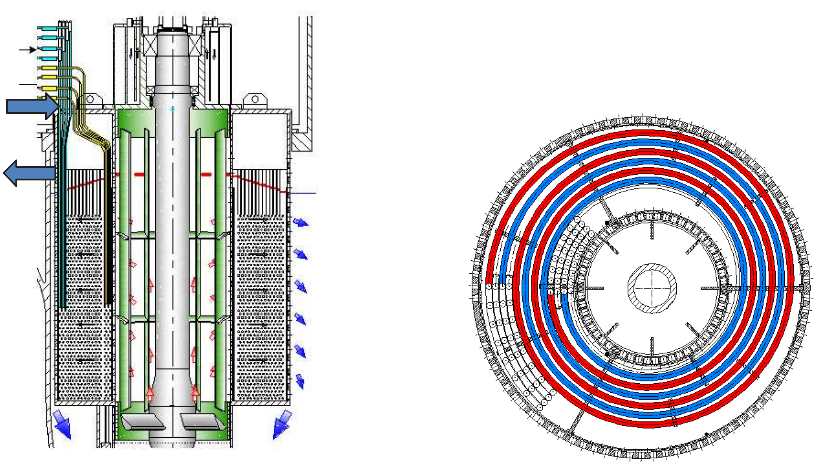


FIG. 8. ELFR spiral tube SG.

The primary pump is an axial flow pump, always running at constant speed, with blade profile designed to achieve the best efficiency. This pump is a high capacity, low head pump, normally designed for flows in excess of 450 m^3/h against heads of 1.5 m or less of lead.

Each pump is installed vertically inside the inner shell of the SG with fluid fed from the bottom. The pump assembly hangs from its support at the top without the shaft's bottom radial support and is removable as a one-piece assembly. The connection to the support is flanged.

The speed of flowing lead relative to the rotating impeller blades is limited by design to about 10 m/s.

The unavoidable high speed of molten lead around and particularly at the tip of the impeller blades, in conjunction with the high temperature, makes the selection of corrosion/erosion resistant construction material a requirement. The most suitable material will be qualified for this application.

The overall feasibility, including manufacture and design qualification of the flat-spiral SG and primary pump units, will be verified by a full experimental programme.

3.2. ELFR DHR system

Two different and independent (physically separated) DHR systems are provided for the ELFR. Each DHR system includes:

(1) DHR1: 4 ICs connected to 4 SGs.
(2) DHR2: 4 ICs connected to 4 dip coolers.

DHR1 and DHR2 meet the requirement of redundancy by means of three out of four loops sufficient to fulfil the DHR safety function. Each DHR can also fulfil the design function of removing the decay heat from the core without exceeding primary lead temperatures even if a single failure occurs.

The DHR1 is the same system installed in ALFRED and is described in Section 2.4.

The DHR2 system (Fig. 9) consists of four independent subsystems, each of them comprising:

- One heat exchanger (IC), constituted by a vertical tube bundle with an upper and lower header.
- One water pool, where the IC is immersed; the amount of water contained in the pool is sufficient to guarantee 3 days of operation.

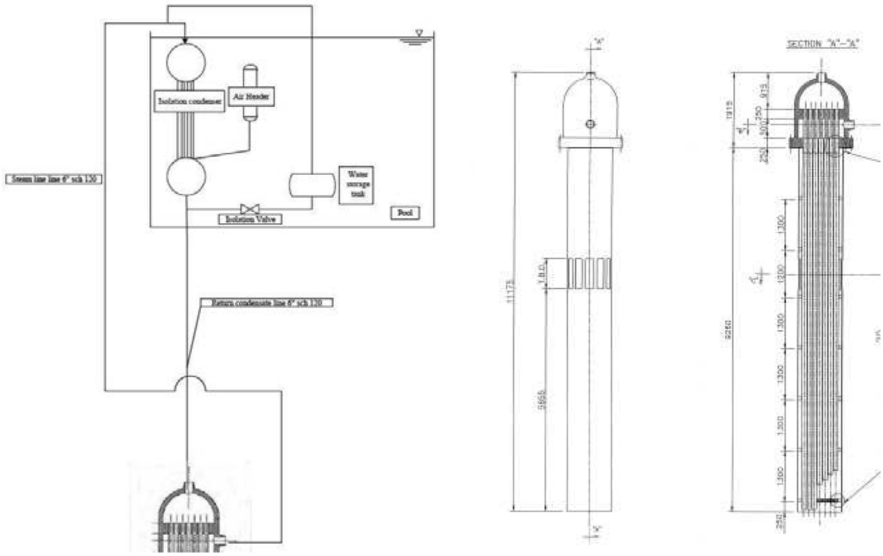


FIG. 9. DHR2 scheme and dip cooler sketch.

- One water storage tank filled with subcooled water at ambient pressure and temperature. The amount of water stored in the water storage tank is defined by design in order to pressurize the DHR2 subsystem as required once it is called to operate.
- One storage tank isolation valve (to meet the single failure criteria this function will be performed at least by two parallel valves).
- One water–lead heat exchanger (dip cooler) constituted by a vertical bayonet tube bundle. The dip cooler is immersed in the reactor downcomer.
- One air header, needed to collect the filler air during the DHR2 operations in order to prevent the accumulation of non-condensable air in the IC tubes.

The dip cooler outlet is connected to the IC upper header through the IC inlet line, while the dip cooler inlet is connected to the IC lower header by means of the IC discharge line.

Under accident conditions, the DHR2 system operates following the reactor trip signal, the main feedwater line isolation signal, the main steam line isolation signal and the DHR1 failure signal. As the DHR2 subsystem actuation signal becomes true, the water isolation valve opens and the water stored in the water storage tank starts to be discharged by gravity into the tube bundle of the dip cooler where it evaporates. The steam produced in the dip cooler condenses in the IC and returns to the dip cooler. The pressure difference between the IC tube bundle, where condensation takes place, and the dip cooler tube bundle, where vaporization takes place, provide the head to establish the water mass flowrate. Preliminary evaluations have shown a grace time of about 6 hours before lead starts to freeze inside the dip cooler. However, to indefinitely increase the grace time, a promising improvement to the DHR2 system is under study.

4. CONCLUSIONS

The paper presents a summary of the LEADER (Lead cooled European Advanced DEmonstration Reactor) project, funded by the European Commission in the frame of the 7th Framework Programme, aimed at the development, to a conceptual level, of an ELFR and of a scaled demonstrator of the LFR technology (ALFRED). In particular, the main design features of both the demonstrator and the ELFR plant are described.

ACKNOWLEDGEMENTS

The authors wish to thank the European Commission for funding the LEADER project in its 7th Framework Programme. Acknowledgment is also due to all the colleagues of the participant organizations for their contributions to many different topics.

REFERENCES

[1] UNITED STATES DEPARTMENT OF ENERGY NUCLEAR ENERGY RESEARCH ADVISORY COMMITTEE, GENERATION IV INTERNATIONAL FORUM, A Technology Roadmap for Generation IV Nuclear Energy Systems, USDOE, Washington, DC (2002).

[2] MANSANI, L., et al., Lead-cooled system design and challenges in the frame of Generation IV international forum, J. Nucl. Mater. **415** (2011) 245–253.

[3] Lead-cooled European Advance Demonstration Reactor, LEADER project (2010) www.leader-FP7.eu

[4] DE BRUYIN, D., et al., “The Fast-spectrum Transmutation Experimental Facility FASTEF: main design achievements (Part 1: Core & Primary System) within the FP7-CDT collaborative project of the European Commission”, Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP), Chicago, IL, 2012, American Nuclear Society, LaGrange Park, IL (2012).

DEVELOPMENT OF FAST SODIUM REACTOR TECHNOLOGY IN THE RUSSIAN FEDERATION

B.A. VASILYEV, D.L. ZVEREV
JSC “Afrikantov OKBM”, Nizhny Novgorod

V.N. YERSHOV
JSC “SPbAEP”, Saint Petersburg

S.G. KALYAKIN, V.M. POPLAVSKIY, V.I. RACHKOV
FSUE “SSC RF IPPE”, Obninsk

O.M. SARAYEV
JSC “Rosenergoatom Concern”, Moscow

Russian Federation

Abstract

The paper provides information about the development of the sodium cooled fast reactors in the former USSR (Russian Federation) starting from the 1950s. The evolution of this technology is traced from the small research reactors to large power units. It is shown how power and parameters were changing in reactor plants; how engineering solutions on the layout, reactor core design, main equipment and systems were evolving; and how the most important problem on increasing the fuel burnup was being gradually solved. Mastering of the dense nitride fuel instead of the MOX fuel is mentioned as an important challenge. Given are operational results for the first power units with the BN-350 and BN-600 reactors; the experience obtained is evaluated. Characterized are the challenges to be faced in the new BN-800 and BN-1200 projects, as well as information about the status of these projects.

1. INTRODUCTION

The Russian Federation (the former USSR) was one of the first countries in the world to start developing fast reactors. With the leadership of A.I. Leipunsky, the first studies in this area were conducted at the Institute of Physics and Power Engineering, Obninsk, Russian Federation. It is according to Leipunsky’s initiative that sodium was used as a coolant in the fast reactor, namely, in the BR-5 reactor design in 1959 [1].

The choice of sodium as the fast reactor coolant has been universally acknowledged and to date, no other fast reactor type has been developed. With reference to the use of liquid metal coolant, it might be noted that ten submarines with lead–bismuth cooled intermediate reactors, which have specific parameters, operation conditions and modes, were developed in the former USSR.

Based on successful experience in the development and operation of the experimental BR-5 reactor, the Russian Federation implemented a substantial R&D programme intended to introduce sodium cooled fast reactors (BN reactors) to the nuclear power structure. In 1969, an experimental reactor BOR-60 was developed [2]. In 1973, the BR-5 reactor was reconstructed with its power uprated from 5 to 10 MW (BR-5 → BR-10). In 1973, the BN-350, the first power reactor prototype, was commissioned [3]. In 1980, the BN-600 demonstration power reactor was commissioned [4]. Currently, construction of the BN-800 reactor is being completed with the purpose of demonstrating closing of the fuel cycle in a fast reactor [5]. Additionally, the BN-1200 reactor design is being developed for commercial construction [6].

The paper characterizes the main results of performed development work and operational experience in the BN power reactors [7]; the trend is shown for the development of design solutions and the fuel cycle of the BN reactors.

2. SELECTING THE POWER AND PARAMETERS OF THE REACTOR PLANT

The thermal power of 1000 MW was selected for the first power reactor BN-350, which corresponds to the mean power level in the first power reactors of other types and in the fossil fuel facilities. Besides, the specific needs of the utility, the Mangyshlak Power Combine, Kazakhstan, were taken into account as it provided electricity and desalinated water to the industry and population of Shevchenko city (now Aktau city, Kazakhstan).

The steam parameters were moderate; the sodium temperature at the reactor outlet was 500°C. The actual thermal power of the BN-350 reactor was limited to 750 MW due to three large inter-circuit leaks in evaporators. Operation was continued after inspection and repair of the evaporators [6].

The thermal power of the BN-600 demonstration reactor was increased to 1470 MW, which in conjunction with the enhanced steam parameters made it possible to obtain the electric power of 600 MW. In the power unit with the BN-600 reactor, three turbines were installed that were commercially available for the fossil power industry with commercially available generators.

In order to ensure standard steam parameters, the sodium temperature at the reactor outlet was adopted equal to 550°C. The BN-600 reactor was constructed within Beloyarskaya NPP Unit 3.

The BN-800 reactor was designed based upon the BN-600 reactor to create a small series of fast reactors in order to ensure transition to the closed fuel cycle with account of the fact that the Russian Federation was developing the RT-1, a spent nuclear fuel reprocessing facility, to reprocess spent fuel from thermal reactors (it has been successfully operating since 1976). The BN-800 design succeeded in substantially increasing the reactor power (up to 2100 MW) as compared to the BN-600 design. Other solutions were adopted that enabled increasing the technical and economic performance. Among other things, the power unit design used the monoblock principle — one turbine, one generator (the rated power of 880 MW). The sodium and steam parameters were preserved at the level of those in the BN-600 reactor plant. Unlike the BN-600 reactor plant, the BN-800 reactor plant uses interim superheating by steam rather than by sodium. Despite the fact that this led to a reduction in the power unit efficiency, it also simplified the steam generator (SG) design, reduced its cost and shortened the length of sodium pipelines.

Construction of the BN-800 reactors was commenced in 1984 at two sites simultaneously — at Beloyarskaya NPP and at a newly developed south Urals NPP. However, the work was halted due to the Chernobyl accident; and only as late as 2006 did the active construction of the BN-800 recommence at Beloyarskaya NPP (as part of Unit 4) according to the design that had been mainly updated in the area of safety enhancement.

Activities to design a large reactor for an extensive commercial construction were commenced in the Russian Federation in 1970 after the BN-600 design had been completed. At that time, a 1600 MW(e) reactor design was initiated that culminated in the design of Unit 5 at Beloyarskaya NPP [8]. In 2002, activities were recommenced to optimize the reactor design for commercial construction. Studies included a 1800 MW(e) reactor [9]. Based on the technical and economic studies, the final design option was adopted with the power of 1200 MW(e) and with the same generator as that in the new Russian WWER reactors (with high speed turbines) [6].

Nevertheless, the major factor for choosing the power was thorough consideration of manufacturing aspects in reactor development, in particular, issues with reactor equipment shipments. In the selected BN-1200 reactor design, all equipment or assembly parts can be shipped by railroad. Note that in addition to the reactor vessel, the large rotating plug will also be fabricated from components at the NPP construction site. The BN-1200 reactor plant parameters have been made higher compared to the previous BN reactor plant designs to enhance the power unit efficiency (see Table 1).

TABLE 1. BN REACTOR PLANT PARAMETERS

	BN-350	BN-600	BN-800	BN-1200
Thermal power (MW)	1000 (750)	1470	2100	2800
Sodium temperature (°C):				
— at reactor inlet/outlet	288/437	368/535	354/547	410/550
— at SG inlet/outlet	415/260	505/318	505/309	527/355
Steam temperature (°C)	405	505	490	510
Steam pressure (MPa)	4.5	14	14	17
Efficiency (net/gross)	*	42.5/40	41.9/38.8	43.5/40.7

* Electricity generation is 125–150 MW and desalinated water production is 100 000 t per day.

3. EVOLUTION OF ENGINEERING SOLUTIONS

3.1. Reactor plant layout

For the BN-350, first power reactor, the distributed layout was used for the reactor plant with six heat removal loops. Note that the power that could be removed by each of the heat removal loops enabled reactor operation at the rated power with five functioning loops. There was no dedicated emergency heat removal system in the reactor plant — the acceptable safety was ensured in the reactor plant by a sufficiently large number of independent secondary loops. Steam from the SG was used for both electricity generation and seawater desalination (the reactor was constructed in the desert on the eastern coast of the Caspian Sea).

In the BN-600, integral layout of the primary circuit is used with three heat removal loops. Each of the three heat removal loops operates with its own turbine and generator. There is no emergency heat removal system in the design, either. In order to enhance safety of the reactor plant when extending the reactor service life beyond 30 years (in 2012), a special emergency heat removal system was developed with a sodium/air heat exchanger connected to one of the secondary loops.

The integral layout of the primary equipment with three heat removal loops was used in the BN-800 reactor; the same as for the BN-600 reactor. The fundamental differences in the BN-800 reactor plant layout are the emergency heat removal system with the sodium/air heat exchanger connected to each of the secondary loops (two heat exchangers in each loop).

For the BN-1200 reactor, the integral layout of the primary equipment with four heat removal loops is used. The monoblock principle has been preserved that was adopted for the BN-800 reactor — one turbine, one generator. The emergency heat removal system is autonomous and consists of four loops and each of them incorporates an in-reactor sodium/sodium heat exchanger and two sodium/air heat exchangers.

The special feature of the BN-1200 reactor is that all the primary systems are fully integrated in the reactor tank, including the sodium purification system and sodium quality control system. Impurities are the most important in terms of reactor safe operation and will be continuously measured by electrochemical detectors (oxygen and hydrogen) and diffusion cell (carbon activity). As a result, the possibility of radioactive primary sodium leakage and fire is excluded. See Table 2 for improved solutions for the BN reactor plant layout.

3.2. Core design and fuel cycle

Russian BN reactor designs were developed to achieve the maximum breeding characteristics — high breeding ratio and short fuel doubling time (T_2). In order to achieve this goal, the reactor core designs provided for a large volumetric fraction of fuel and high power density. As soon as the BN reactors were being practically mastered, a tendency arose to reduce the power density in the cores with account of actual fuel cycle economics and estimation of changes in it. In the BN-1200, the core power density is reduced by approximately half compared to the BN-600 and BN-800 reactors, which ensured a respective increase in the fuel assembly (FA) life at the same fuel burnup. The BN reactor core parameters are shown in Table 3; and the residence time for FA as a function of core power density is shown in Fig. 1.

TABLE 2. IMPROVING THE SOLUTIONS FOR THE BN REACTOR PLANT LAYOUT

	BN-350	BN-600	BN-800	BN-1200
Primary circuit layout	Distributed	Integral	Integral	Integral
Auxiliary sodium systems outside the reactor vessel	Yes	Yes	Yes	No
Number of heat removal loops	6	3	3	4
Number of turbines and generators	4 + 4	3 + 3	1 + 1	1 + 1
Presence and design of the emergency heat removal system	No	One channel with a Na/air HX connected to the secondary circuit	Three channels with Na/air HXs connected to the secondary circuit loops	Four channels with a Na/air HXs connected to autonomous primary Na/Na HXs
Number of SG modules	6 × 2	3 × 24	3 × 20	4 × 2

TABLE 3. SOME OF THE BN REACTOR CORE PARAMETERS

	BN-350	BN-600	BN-800	BN-1200
Fuel volume fraction	0.380	0.443	0.429	0.471
Core height (m)	1.06 (1.03)	0.75 (1.03)	0.90	0.85
Fuel rod diameter (mm)	6.1 (6.9)	6.9	6.9	9.3
Average power density (kW/L)	(390)	(400)	450	230
Fuel type	UO_2	UO_2	MOX	Nitride/MOX

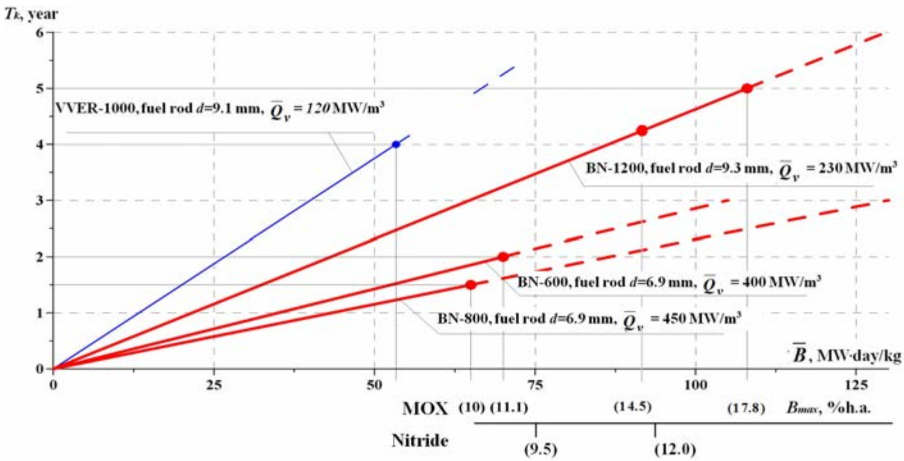


FIG. 1. BN-1200 residence time for FA duration as a function of fuel burnup.

The core height in BN designs is limited to the value that is a little greater than one metre in the BN-350 and BN-600 reactors with enriched uranium fuel and less than one metre for cores in the BN-800 and BN-1200 that are designed for the MOX fuel. Largely, this limit is due to the approach employed in the Russian Federation to limiting the sodium void effect to less than β_{eff}. Note that in the BN-800 and BN-1200 reactors with MOX fuel, the core design is adopted with an upper sodium cavity and upper axial absorbing screen [10].

The special feature of the BN-1200 reactor core is that its fuel has uniform enrichment and the burnup of FAs to be unloaded is equalized by the different times of their core residence.

Along with the sodium coolant for fast reactors, Leipunsky suggested using ceramic fuel in the form of dioxide that is characterized by high thermal stability and high melting temperature [11]. In order to simplify the task and to master the BN reactor technology, enriched uranium fuel was adopted for the BN-350 prototype reactor and the BN-600 demonstration reactor; and it was justified under the conditions with a well-developed uranium industry in the Russian Federation.

Substantial experience in improving the fuel rod design and FA was gained in the course of operating the BN-350 and BN-600 to increase the fuel burnup. This experience is versatile because the basic factors critical for the lifetime, first of all the radiation damage to structural materials, do not depend on the oxide fuel type. The burnup that has been currently achieved in the uranium oxide fuel in the BN-600 reactor is ~70 MW×d/kg (more than 11% h.a. by the maximum value); it is much greater than that in the WWER reactors (~50 MW × d/kg) [12].

With the use of the BOR-60, BN-350 and BN-600 reactors, the MOX fuel technology was tested in two options, namely, MOX pellet fuel (similar to the uranium dioxide fuel) and vibropacked MOX fuel [13].

Presently, a fuel batch is being fabricated for the first BN-800 core that consists of FAs with uranium oxide fuel; FAs with MOX pellet fuel; and FAs with vibropacked MOX fuel. FAs with the MOX fuel are being fabricated in experimental production facilities. Simultaneously, industrial production facilities are being developed to provide that the BN-800 is fully loaded with the MOX pellet fuel through gradually replacing the FAs of the initial fuel batch in the reactor [14, 15].

Provided that ChS-68 (16Cr15Ni2Mo2MnTiB) cold worked austenitic steel proven in the BN-600 reactor is used for fuel cladding, the planned MOX fuel burnup in the BN-800 reactor would be ~10% h.a. by the maximum value. It could be up to ~12% h.a. provided that EK-164 cold worked austenitic steel (16Cr19Ni2Mo2MnNbTiVB) is used that is undergoing testing (there is positive experience with the uranium fuel up to the burnup of ~13.5% h.a.) [16].

A special feature of BN-800 operation will be the use of MOX fuel based on the weapons grade plutonium to dispose of it. Simultaneously, for the purposes of closing the fuel cycle, MOX fuel will be used that is based upon plutonium from WWER reactors; and then, based upon the BN-generated plutonium.

For the BN-1200 reactor, cores are being developed with two types of fuel: MOX and nitride fuel. The goal of introducing the nitride fuel is to fully meet the requirements for the inherent safety of the Russian “Breakthrough” project [17]. Actions are taken to accelerate acquisition of experimental data that should ensure the possibility of using nitride fuel, starting with the first loading in the BN-1200 reactor. The MOX fuel is being developed as a backup option for the case of difficulties and delays in validation of the nitride fuel.

Fuel could be produced and reprocessed in both centralized facilities and NPP in-house production facilities that should be suitable for both MOX and nitride fuels.

In any option of the fuel cycle, it should be ensured that BN-generated minor actinides are burned.

The basic design mode of the core is operation with the refuelling interval of 1 year (330 EFPD) with the four-year life of the bulk FAs to the average burnup of ~90 MW·d/kg for the MOX fuel; and ~70 MW·d/kg for the nitride fuel.

Advanced ferritic-martensitic steels EK-181 (12Cr2WVTaB) and ChS-139 (12CrNiMoWNbVB) are considered as a fuel cladding material (up to date, only the EP-450 steel has been tested that is successfully used for FA wrapper tubes). In order to achieve the goal of increasing the burnup, work is in progress to develop dispersion hardened steels [18]. For the initial phase in operation of the pilot BN-1200 reactor, one option is considered with operation at a reduced burnup and using the EK-164 cold worked steel for fuel cladding.

3.3. Reactor design

Largely, the reactor determines the performance, safety and material consumption of a reactor plant. The integral BN reactor concept was developed in the design of the BN-600 reactor plant and applied without major changes in the BN-800 and BN-1200 reactor plant designs. The features of this concept are that the reactor vessel is supported at its lower part; there is the upper deck in the form of a tapered roof and rotating plugs; there is a box-like supporting structure for the core, IHXs and MCPs that ensures sodium circulation in individual IHX-MCP loops and there are check valves on MCPs that enable isolation of any heat removal loop.

The main changes in the BN-800 reactor design compared to the BN-600 design are that a welded joint is used for the reactor support instead of roller supports, which enhances the reactor seismic resistance; and that the core catcher is introduced to hold the corium in the lower part of the reactor.

In addition to the changes related to the full integration of the primary equipment in the BN-1200 reactor design, the in-vessel shielding has been simplified — it is basically made of boron carbide and concentrated immediately around the core. At the same time, the neutron flux towards the barrel that forms the bulk of FAs in the core has been reduced. This removed the reactor life limitation due to barrel shape changes that had been predicted for the BN-600 and BN-800 reactors.

3.4. SG design

Another important structural element that determines reactor plant performance is an SG. As a result of problems with SGs during the initial phase in operation of the BN-350 reactor plant, a sectional-modular SG design concept was selected for the BN-600 reactor, which ensured isolation of SG modules having possible inter-circuit leakage without reducing the reactor plant power.

This concept with certain changes in the SG design is used in the BN-800 reactor plant. On the one hand, the experience gained in operation of BN-600 SGs verified that the sectional-modular SG concept is efficient in terms of ensuring reliable operation of the reactor plant; and on the other hand, it showed a trend in developing a sufficiently reliable design for a large module SG, which became a basis in the development of the BN-1200 reactor plant. In the BN-1200 reactor, a straight-tube SG design is used with two modules per loop, either of which joins the evaporator and superheater portions of the tubing system.

Figure 2 shows one of eight SG sections in a secondary circuit loop.

3.5. Refuelling system

Another large structural system that ensures reactor operation is the refuelling system. When the BN-600 reactor was being developed, a sufficiently effective refuelling system was developed that ensured start of refuelling shortly after reactor shutdown (2–3 days after the shutdown). The refuelling operation rate makes it possible to do refuelling in less than 10 days. The refuelling system that had been developed for the BN-600 reactor was used without major changes to the BN-800 reactor plant. As for the BN-1200, significant changes have been made to the refueling system design to reduce its material consumption. The main new solution is that the sodium buffer store, which is used for interim storage of FAs before their washing, has been excluded from the refuelling system [19]. It has been achieved through sufficiently long (two years) holding of spent FAs in in-reactor storage.

4. REQUIREMENTS FOR NEW BN REACTOR DESIGNS

Development of the BN-800 reactor meets a whole range of goals to master the sodium cooled fast reactor technology. First, the knowledge and skills that have been obtained through development and operation of the BN-600 reactor are used in practice at a new level. Second, the MOX fuel and then other elements of the fuel cycle will be mastered. Third, the entire infrastructure required for developing a commercial BN reactor will be maintained and updated.

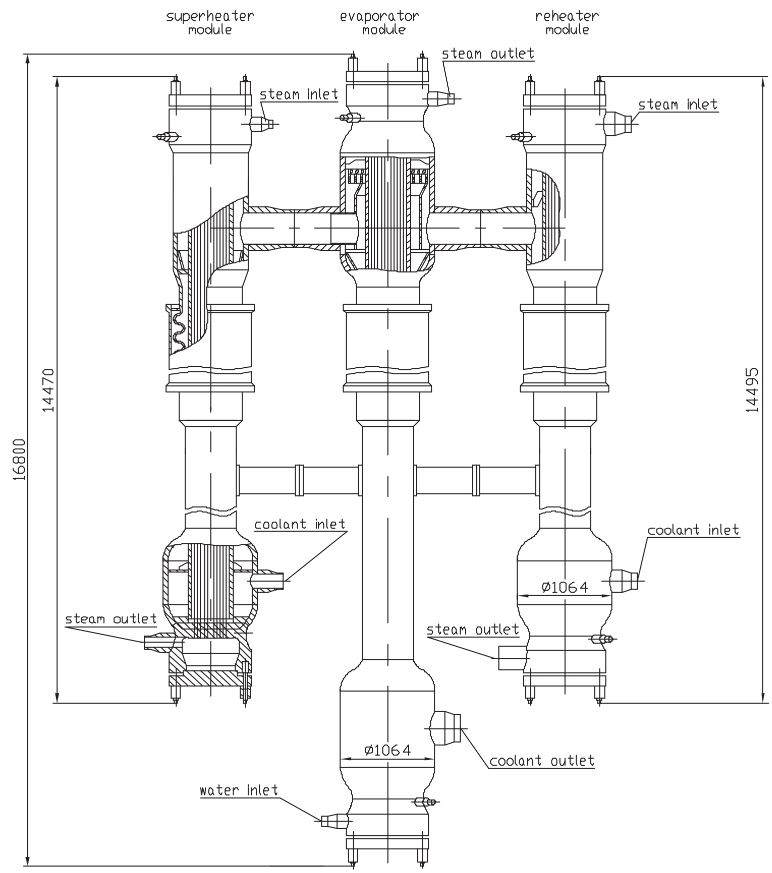


FIG. 2. SG section.

Especially important is the focus on safety of the BN-800 reactor of which construction was interrupted after the Chernobyl accident. The BN-800 design was so well developed that the first licence after the Chernobyl accident was given for its implementation (construction recommencement).

The BN-1200 is being developed for serial construction. It is not only that the BN-1200 design should ensure breeding of fuel and burning of minor actinides. The reactor should have a high safety level and be economically competitive to other types of energy source. As a safety criterion, there is a requirement for the reactor that the need should be prevented to evacuate population living in the vicinity of an NPP. This requirement should be met in all technically possible accidents. Such accidents also include extremely unlikely ones in which initial events are superimposed by failures of all active safety systems and single failures in passive safety systems provided in the design to limit the consequences of such accidents. In order to achieve the respective safety level, the inherent safety properties are developed to the maximum and new passive protective devices are used in the BN-1200 design: absorber rods with temperature triggered action and an above-reactor leaktight room to receive gaseous radionuclides and aerosols during accidents with significant increase of temperature and pressure in the reactor [17].

As for ensuring competitiveness of the BN-1200 power unit, this is achieved through substantial simplification of the BN-1200 reactor plant design compared to the BN-800 design and through the use of new solutions in the power unit design [20]. The effectiveness of new solutions in terms of reducing the material consumptions and eventually the reactor plant cost is shown in Fig. 3. It is important in terms of achieving high technical and economic performance that confidence is gained in the feasibility of designing the power unit for its 60-year service life [21].

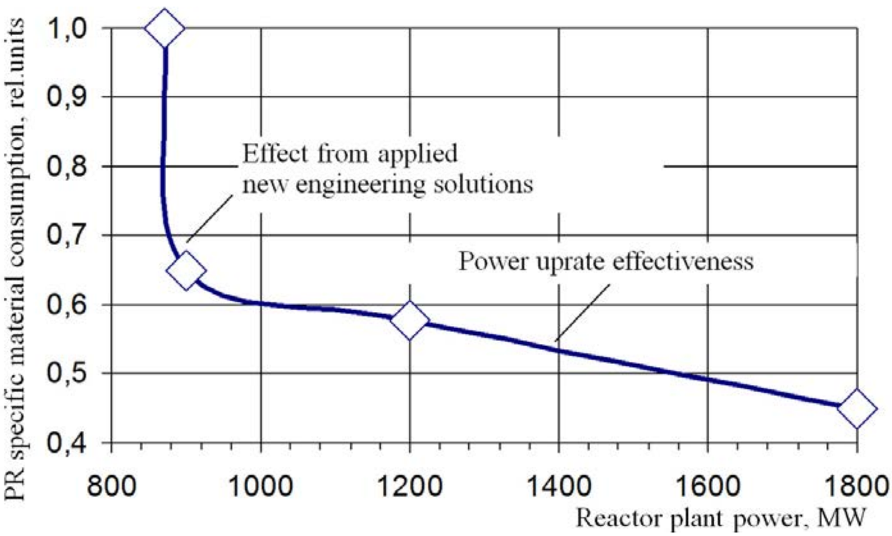


FIG. 3. Reduction in material consumption in the BN reactor plants.

5. STATUS OF PROJECTS

The BN-350 reactor was successfully operating from 1973 to 1999 [22]. After Kazakhstan was established as an independent State, the reactor was shut down basically due to organizational problems with settlement of issues with regard to designers' field supervision providing reliability and safety for the reactor after its 20-year design service life had expired. A large scope of work has been done to decommission the reactor, including disposal of sodium [23].

The BN-600 is currently the only fast power reactor operating in the world. From the moment the reactor was commissioned in 1980, it has been in quite successful operation for 32 years (see Table 4) [12, 24, 25].

TABLE 4. MAIN RESULTS OF BN-600 OPERATION

Power factor	77.2% over the period 2001–2011 with the unplanned losses in the power factor not exceeding 0.7%. The planned time for the scheduled preventive repair and refuelling are determined by two shutdowns per year and a repair of three turbo-machineries
Sodium leaks	27 outward sodium leaks (with 5 radioactive sodium leaks out of them) and 12 SG leaks. The outward leaks were basically due to deviations in fabrication quality of auxiliary pipelines. The last outward sodium leak was in 1993; SG leak in 1991
Fuel burnup and FA residence time	The average fuel burnup has been increased from 42 to 70 MW×d/kg; the FA residence time by ~2 times
Emergency shutdowns of the reactor	The average number of emergency reactor shutdowns is ~0.2 over 7000 hours in operation (for NPPs in the world, it is ~0.6) over the period 1990–2011; there has been no emergency reactor shutdowns over the period 2001–2011
Average radioactive gas emissions over the period 2001–2011	<1% of the allowable level
Collective dose rate for personnel over the period 2001–2011	~0.475 person Sv per year

The BN-600 reactor plant construction has preserved high serviceability, which made it possible to validate the possibility of extending the power unit service life to 45 years. In April 2010, Gostekhnadzor, of the Russian Federation, issued a licence for extending the reactor service life for 10 years on condition that a number of power unit equipment items would be replaced and certain measures would be taken to enhance reactor safety (mainly, it is the development of the additional emergency heat removal system).

According to the results of the additional safety analysis (stress tests) for the BN-600 reactor after the accident at the Fukushima NPP, sufficiently high reactor resistance to internal and external events was verified, including preservation of the safety functions in earthquakes of 7-point intensity (that is 1 point higher than the maximum design earthquake).

Additionally, it is planned to equip the power unit with an autonomous portable 2 MW diesel generator station for emergency power supply to critical mechanisms, as well as with a 0.2 MW station for emergency power supply to instrumentation switchboards and a reserve control panel [24].

Construction of the BN-800 reactor at Beloyarskaya NPP is entering its final phase [16]. The major part of power unit equipment has been delivered to the construction site. Almost all the equipment is manufactured by Russian factories. The construction of the BN-800 reactor utilizes a new approach to final assembly of the reactor vessel from elements supplied to the NPP site. When the BN-600 was being constructed, the final assembly of the reactor vessel was performed in the central hall. Unlike that, a special workshop was erected for the BN-800 reactor to assemble large preassembled modules that were installed in the reactor well without completing the construction work on reactor building structures. The gained experience that turned out to be a successful experience is important in terms of ensuring accelerated construction of commercial BN reactors. At the present time, after the hydraulic testing has been completed on the reactor vessel, activities are being completed to install reactor internals, whereupon installation work will be commenced on reactor equipment. Work is also in progress to prepare the turbine and generator for installation. The reactor startup is planned for 2014.

Currently, the detailed design of the reactor plant is being developed for the BN-1200 project, as well as the detailed design of the turbine plant. The completion of the final reactor plant design with validation of engineering solutions and with results of current R&D work is planned for 2014. It is planned to fully complete the R&D work by 2016 except for the work on ensuring high fuel burnup. In 2014, main design solutions for the power unit will have to be selected and validated. The issue of constructing the first BN-1200 reactor will be resolved after the feasibility of achieving the target economics, safety and environment protection is verified in the course of designing.

6. CONCLUSION

Over the period from 1950s, the Russian Federation has gained substantial experience in developing and mastering of sodium cooled fast reactors. The current period is characterized by dynamic activities to further promote reactors of this type. Construction of the BN-800 reactor and designing of the BN-1200 reactor must become a basis for starting commercial production of the BN reactors and closing the fuel cycle.

REFERENCES

[1] BAGDASAROV, Yu.E., et al., Technical issues with fast reactors, Atomizdat, Moscow (1967).

[2] Machine Building Encyclopedia, Vol. IV-25 Nuclear machine building. M.: Mashinostroyeniya, 2005, p. 710.

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Progress in Liquid Metal Fast Reactor Technology, IAEA-TECDOC-876, IAEA, Vienna (1996).

[4] [4] OSHKANOV, N.N., SARAYEV, O.M., BAKANOV, M.V., et al., Thirty-year experience in operation of the BN-600 sodium-cooled fast reactor, Atomnaya Energiya **108** (2010) 186–191.

[5] SARAYEV, O.M., NOSKOV, Yu.V., ZVEREV, D.L., et al., BN-800 design validation and status of power unit construction, Atomnaya Energiya **108** (2010) 197–201.

[6] RACHKOV, V.I., POPLAVSKIY, V.M., TSIBULYA, A.M., BAGDASAROV, Yu.E., et al., Concept of the advanced power unit with a sodium-cooled fast reactor, Atomnaya Energiya, **108** (2010) 201–206.

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Unusual Occurrences during LMFR Operation, IAEA-TECDOC-1180, IAEA, Vienna (2000).

[8] KOSTIN, V.I., VASILYEV, B.A., The purpose of constructing the BN-800 and possibilities of developing advanced fast reactors, Atomnaya Energiya **102** (2007) 21–26.

[9] VASILYEV, B.A., BUDILSKY, A.D., KAMANIN, Y.L., et al., BN-1800: a next generation fast breeder, J. Nucl. Eng. Int. **49** (2004) 20–24.

[10] POPLAVSKIY, V.M., TSIBULYA, A.M., HOMYAKOV, Yu.P., "Reactor cores and fuel cycle for the advanced sodium cooled fast reactor", Proc. Int. Conf. on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), Kyoto, Japan, 2009, IAEA, Vienna (2012).

[11] LEIPUNSKY, A.P., et al., Proceedings of the Second International Conference for Peaceful Use of Nuclear Energy, Geneva (1958), Papers by soviet scientists, Atomizdat **2** (1959) 215.

[12] OSHKANOV, N.N., BAKANOV, M.V., POTAPOV, O.A., Operational experience in the BN-600 power unit at Beloyarskaya NPP, Atomnaya Energiya **96** (2004) 201–206.

[13] POPLAVSKY, V.M., ZABUDKO, L.M., SHKABURA, I.A., et al. “Fuels for advanced sodium cooled fast reactors in the Russian Federation: state of the art and prospects”, Proc. Int. Conf. on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), Kyoto, Japan, 2009, IAEA, Vienna (2012).
[14] KISLY, V.A., SHISHALOV, O.V., KORMILITSYN, M.V., “Vibropac MOX-Fuel for Fast Reactors”, paper presented at Int. Conf. GLOBAL: Toward and Over the Fukushima Daiichi Accident, Makuhari, Japan, 2011.
[15] GLAZOV, A.G., GORBACHEV, M.K, Developing industrial production of MOX fuel, J. Nucl. Environ. Saf. **1** (2012) 20–21.
[16] ZVEREV, D.L., VASILYEV, B.A., SEDAKOV, V.Yu., et al.,”Mastering the sodium-cooled fast reactor technology. Development of the BN-800 reactor plant”, Conference BN-600 — 30 Years, Zarechny, Sverdlovskaya (2010).
[17] VASILYEV, B.A., YEVSEYEV, V.I., KUZAVKOV, N.G., et al., Implementing the inherent safety principle in the BN-1200 reactor plant design, J. Nucl. Environ. Saf. **1** (2012) 62–65.
[18] SHKABURA, I., Time for the fast ones, Rosenergoatom **12** (2009) 6–11.
[19] VASILYEV, B.A., TIMOFEYEV, A.V., LYUBIMOV, M.A., BUDYLSKY, A.D., (JSC “Afrikantov OKBM”), Layout and design solutions for the refueling system in the advanced BN reactor, Atomnaya Energiya **108** (2010) 342–346.
[20] ZVEREV D.L., VASILYEV B.A., “BN reactor technology. Development status”, 7th Int. Scientific and Technical Conf. Safety, Efficiency and Economics of Nuclear Power, Moscow, Russian Federation, 2010.
[21] VASILYEV, B.A., VILENSKY, O.Yu., KAYDALOV, V.B., these Proceedings.
[22] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of liquid metal cooled fast reactor technology, IAEA-TECDOC-1083, IAEA, Vienna (1999).
[23] SHIROBOKOV, I.P., VASILYEV, I.I., VALDAYEV, I.V., Bringing the BN-350 reactor to the safe storage condition, J. Nucl. Environ. Saf. **3** (2011) 124–127.
[24] BAKANOV, M.V., NOSOV, Yu.V., POTAPOV, O.A., “Operational experience in the BN-600 sodium-cooled fast reactor”, paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.
[25] BAKANOV, M.V., POTAPOV, O.A., “30-year Commercial Operating Experience from the BN-600 Reactor”, Proc. of High Schools, J. Nucl. Power **1** (2011) 22–26.

MANUFACTURE AND ERECTION OF SFR COMPONENTS: FEEDBACK FROM PFBR EXPERIENCE

P. CHELLAPANDI, S.C. CHETAL
Indira Gandhi Centre for Atomic Research,
Kalpakkam, India

Abstract

Manufacture of large size thin walled vessels, made of stainless steel with tight form tolerances, machining and assembly of grid plate, and steam generators with close tolerances, are some of the challenging issues which have been successfully resolved for the 500 MW(e) Prototype Fast Breeder Reactor (PFBR). In order to build this unique capability in the Indian industries and also assess manufacturing tolerances that can be achieved by these industries, elaborate manufacturing technology development works were undertaken prior to the start of construction of the PFBR. The welding and pre-service inspection techniques and strategies were finalized through many successful mock-up trials. Judicious choice of tolerances, number and location of welds and inspections had been made. Robust criteria have been applied for the acceptance of manufacturing deviations and material compositions. Indigenous materials were used after qualifications of manufacturing processes of direct relevance, apart from routine standards. From the rich experience gained through the manufacture and erection of reactor assembly components of the PFBR, important guidelines and approaches were derived. The history of design, manufacturing and erection experiences is well documented to develop future applications aimed at achieving economic competitiveness of the technology with respect to the enhanced safety requirements. These are very useful for the success of future nuclear reactors planned by the Department of Atomic Energy.

1. INTRODUCTION

A 500 MW(e) Prototype Fast Breeder Reactor (PFBR), a typical pool type sodium cooled fast reactor (SFR), designed and developed at the Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam in India is in the commissioning stage. The PFBR is a pool type reactor with two primary and two secondary loops with 4 steam generators 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 hot pool, after transporting its heat to four intermediate heat exchangers (IHX), mixes with the cold pool. The circulation of sodium from the cold pool to the hot pool is maintained by two primary sodium pumps and the flow of sodium through the IHX is driven by a level difference

(1.5 m of sodium) between the hot and cold pools. The heat from the IHX is in turn transported to eight steam generators (SG) by sodium flowing in the secondary circuit. Steam produced in the SG is supplied to the turbogenerator. The main vessel in the reactor assembly houses the entire primary sodium circuit, including the core. Sodium is filled in the main vessel with free surfaces blanketed by argon. The inner vessel separates the hot and cold sodium pools. The reactor core consists of about 1757 subassemblies, including 181 fuel subassemblies. The control plug, positioned just above the core, houses the 12 absorber rod drive mechanisms. The top shield supports the primary sodium pumps, IHX, control plug and fuel handling systems. The PFBR uses mixed oxide with both natural and depleted uranium, 21% Pu oxide in the inner core and 28% Pu oxide in the outer core. For the core components, 20% of cold worked D9 material (15% Cr–15% Ni with Mo and Ti) is used to obtain 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 SG. The PFBR is designed for a plant life of 40 years with a load factor of 75%. The commissioning of the PFBR has already been started.

Bharatiya Nabhikiya Vidyut Nigam Limited (BHAVINI), a Government company, is constructing the PFBR, India's first commercial SFR project, in coordination with the Indira Gandhi Centre for Atomic Research, which is responsible for the design and R&D. Beyond the PFBR, the Department of Atomic Energy is planning to build a series of 500 MW(e) SFRs adopting the twin unit concept. The first twin unit (2 × 500 MW(e)) would be constructed at Kalpakkam near to the PFBR. The capital cost of these twin units, as well as construction time, should be reduced significantly for the commercial exploitation. To achieve this, many features have been identified. One of the essential steps is incorporation of lessons learned during manufacturing and erection of the PFBR components. This paper provides the lessons learned from the significant efforts put forth for the successful manufacturing and erection of reactor assembly components.

2. SALIENT FEATURES OF THE SFR COMPONENTS WITH REFERENCE TO CONSTRUCTION

SFR components, in general, are characterized as large diameter thin walled shell and slender structures. Stringent manufacturing tolerances are specified to enhance their buckling strength as well as supporting the minimum possible vessel dimensions. In the reactor assembly, and main vessel, thermal baffles, inner vessel, core support structure and grid plate are to be positioned sequentially, maintaining the coaxial configuration with the safety vessel, so that the core central line will be in-line with the central lines of the vessels. This is one of the requirements to facilitate smooth operation of control rods as well as for facilitating accurate monitoring of temperatures of sodium emerging from the core subassemblies. Further, they have to be erected accurately to maintain the annular gaps for uniform sodium flows and temperatures. During the manufacturing stage, single side welds are unavoidable at some difficult locations particularly in the case of box type structures. In-service inspection is difficult with the presence of sodium and, hence, stringent quality control is required in the pre-service level itself. From the dimensional stability point of view, residual stresses should be kept minimal by adopting robust heat treatment processes and mock-up trials. It is preferable to use a minimum number of materials from the consideration of economy and material data generations. This also enhances reliability of performance of materials during operation. Austenitic stainless steels, the main structural material, in particular call for careful consideration for welding without significant weld repairs and distortions. International construction experience and that gained with the PFBR indicate that reactor assembly components determine the project time schedule, even though their cost is relatively small compared to civil sodium circuits and balance of plant (BoP). There is only limited experience in manufacturing and erection of components. These factors apart, the design and manufacturing codes are still evolving. These are the major challenges in the manufacturing and erection of reactor assembly components. Figure 1 depicts the arrangement of reactor assembly components for the PFBR.

3. MANUFACTURING STRATEGY FOLLOWED

While IGCAR designed the reactor assembly components, six major industries in the country were involved in the manufacture (Fig. 2). It has been found that industries request the designers to accept minor deviations in the composition of materials already available to them. Due considerations were given for such requests with accumulated knowledge/expertise over the years. This has avoided procurement of fresh materials. Thus, as far as possible, fresh raw material procurement, including weld electrode and filler wire (thickness of spherical header) was avoided by using suitable alternatives. This has resulted in significant savings in project schedule. Manufacturing deviations were accepted in the material properties based upon component duty and environment, for example, Mo content in the weld electrode for low temperature application and ductility requirement for tools. Certain deviations from manufacturing and welding procedure qualifications were accepted, e.g. relaxation of requirements for non-load carrying and non-sodium wetted boundaries of the safety vessel and core catcher. Testing and inspection procedures were simplified based on their

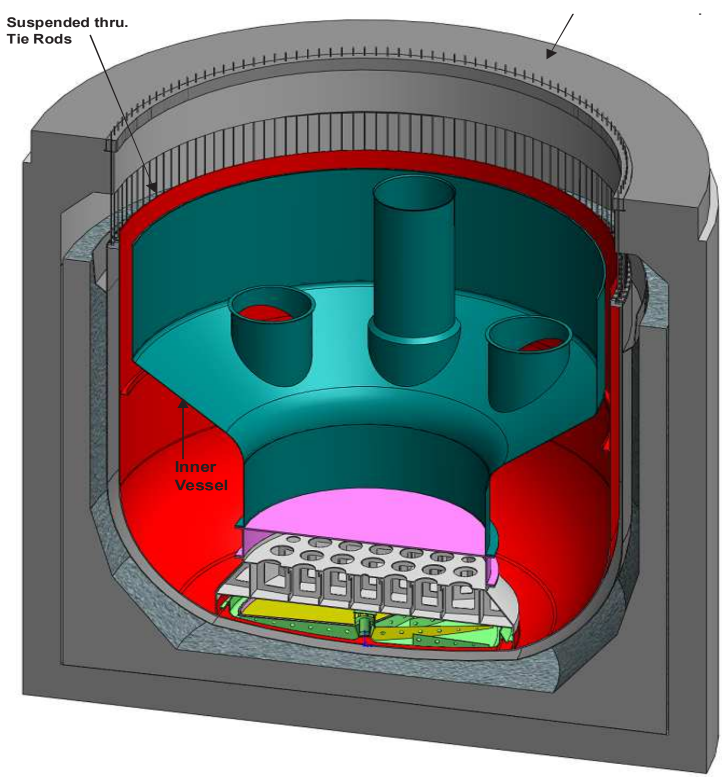
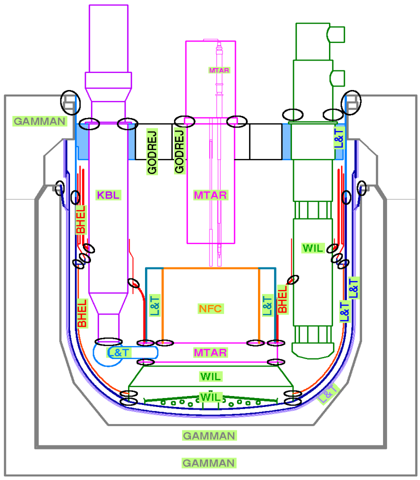


FIG. 1. PFBR reactor assembly components.

importance and duty. An alternative bend test procedure was suggested for the dissimilar welds at the roof slab. Parts assembly procedures were simplified to achieve the tight requirement of erection tolerances. However, for the main vessel, care has been taken so that there is practically no repair of welds and laminations. With thorough and comprehensive analysis respecting interface requirements with other components, as-built dimensions were accepted. For certain non-critical cases (e.g. core catcher and safety vessels), the concept of a 'fit for purpose' approach was adopted judiciously.

4. MANUFACTURING CHALLENGES

For the manufacture of components, robust construction codes, standards and methodologies should be employed. The most important aspect is specification of manufacturing tolerances, which should be written based on a rational basis of functional requirements and structural integrity considerations.



Sl.no	Components	Industries
1	Core Subassemblies	NFC, Hydrabad & L&T Hazira
2	CSRDM & DSRDM	MTAR, Hydrabad
3	Safety Vessel	L&T (SAS with KRR petals)
4	Main vessel	L&T (SAS with KRR petals)
5	Thermal Baffles inc. cooling pipe	BHEL, Trichy
6	Inner Vessel	BHEL, Trichy
7	Core Catcher	WIL, Walchandnagar
8	Core Support Structure	WIL,Walchandnagar
9	Grid Plate	MTAR (Hardfacing by OMPLAS)
10	Primary Pipe	L&T Powai
11	Roof slab	L&T Hazira
12	LRP & SRP	Godrej
13	Control plug	MTAR, Hyderabad

FIG. 2. Major industries involved in the manufacture of reactor assembly components of the PFBR.

The manufacturing tolerances should truly reflect similar experiences and industrial capability. For those items, which the industry is attempting for the first time, appropriate and novel mock-up trials are essential. The assembly sequences should be well understood and efficient handling schemes are to be designed and validated. It is essential that the components should be manufactured in a practical manner with insignificant repairs. A few challenging issues experienced during manufacture of PFBR components are addressed here.

4.1. Manufacture of large diameter thin shell structures

For the large diameter thin vessels, i.e. main vessel, thermal baffles, inner vessel, safety vessel, the major manufacturing challenges are: the basic plates should not have any defects such as laminations (high quality control is essential), long lengths of welds while integration of individual petals, stringent control on the manufacturing deviations, such as form tolerances (<½ thickness), verticality and horizontality (<±2 mm), high quality welds and low residual stress to be achieved without any heat treatment. High quality welds achieved with insignificant weld repairs, cold forming limited to <10% and close form tolerances of ±12 mm (≤0.2% R) are the major achievements. These are achieved by adopting stringent dimensional control on petal level and subsequently robust weld fit up and weld sequence methodology, state-of-the-art techniques for inspection and quality control, numerical simulations of forming and welding procedures, innovative mock-up trials, lessons learned from the experiences of various industries and elaborate technology development exercises. Figure 3 depicts form tolerances achieved in the thin shells.

4.2. Manufacture of grid plate

The grid plate performs many important functions: to support the core subassemblies (SA), the inner vessel and in-vessel transfer position, as well as to serve as a plenum for distributing the coolant flow to the SA and main vessel cooling. Geometrically, the grid plate is a circular box type structure of bolted construction. It consists of upper and lower plates interconnected by a number of tubes called ‘sleeves’ and an outer cylindrical shell. The SA rests on the conically shaped hard faced surface (chrome nitride) at the top of the sleeve. The nickel based alloy chosen for hard facing is highly susceptible to cracking and it was required to deposit this alloy on components of very large dimensions without any cracks. Often, the volume of deposition is so high that obtaining deposits without cracking required special technology development efforts involving designers, fabricators and metallurgists. Through the technology development effort, the hard facing of the bottom plate of the prototype grid plate assembly was successfully completed. Further, the hard facing of the inner surfaces of a very large number of sleeves (about 1757) in which the fuel subassembly rests was another critical task that was successfully completed through indigenous technology. Nozzle welding with least distortion, large diameter colmonoy hard

Component	ASME	Form Tolerance on Radius (mm)		
	(ID_{Max} - ID_{Min})	RCC-MR	PFBR	
			Specified	Achieved
Main Vessel	± 70	± 50	± 12	< ± 12 (during fit up) < ± 18 (at isolated locations)
Safety Vessel	± 70	± 50	± 12	± 12 (at majority locations) ± 18 (at isolated locations)
Inner Vessel	± 67	± 50	± 12	< ± 8 (during fit up) < ± 20 (at isolated locations)

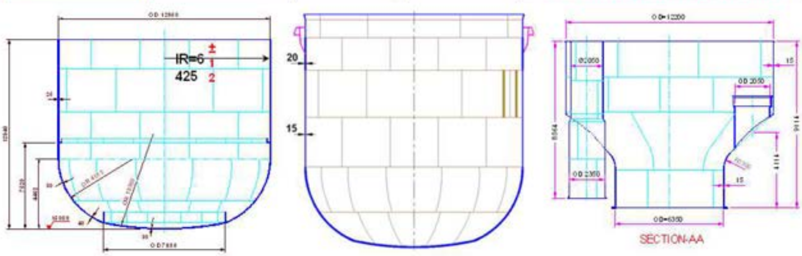


FIG. 3. Form tolerances achieved in the thin shells of the PFBR.

facing without defects, hard facing of a large number of sleeves on the inner diameter at a depth of 500 mm are some of the achievements accomplished during the manufacture of the grid plate. These are achieved by adopting challenging methods in dimensional control, innovative distortion control methods, modern dimensional inspection methods, handling of plates with a large percentage of perforation, an innovative way of heat treatment of large dimensioned components and novel techniques indigenously developed for handling/assembly of large numbers of items (>14 000). Figure 4 shows various manufacturing stages of the grid plate.

4.3. Manufacture of roof slab

A box-type structure, made of A48P2 carbon steel (CS) (similar to ASTM A516 Grade 70 with a lower limits for sulphur) specified in the French Nuclear Code for FBRs RCC-MR, consists of a bottom plate and a top plate with the space between them being partially filled with concrete. This box-type structure is a massive structure of 12.9 m diameter, 1.8 m height and weighs about 650 t. The thickness of the plate was chosen to be 30 mm from the consideration that stress relieving heat treatment is not mandatory for this thickness as per various fabrication codes. The fabrication of the box-type structure involves a dissimilar metal weld joint between A48P2 CS and 316LN SS plates at the bottom of the structure. A specific issue causing significant time delays are laminar tear problems. The plates were to meet through thickness ductility of 25% and ultrasonic inspection requirements. However, during fabrication of the roof slab, lamellar tearing was observed in one of the heats of this steel, though this heat had met all the specification requirements. This was overcome by buttering the plate surface by welding, as permitted in ASME, and by changing the joint design. Figure 5 gives the details of the various weld configurations in the roof slab and the configurations selected for avoiding laminar tearing based on extensive R&D activities.

FIG. 4. Manufacturing stages of grid plate.

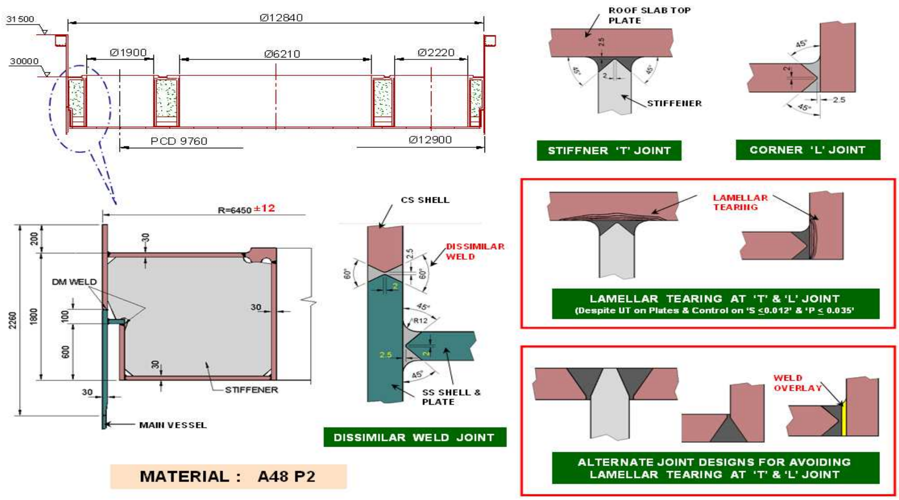


FIG. 5. Weld configurations to overcome laminar tear problems.

4.4. Manufacture of SGs made of 9Cr-1Mo

The SGs are among the most critical components of the PFBR because their reliable operation determines the load factor of a SFR plant. The SG is made of modified 9Cr 1Mo steel. It is a tall component of about 25 m in height and has more than 500 tubes. The tube-to-tubesheet joint is the most critical one, since any leak in the tube-to-tubesheet joint can result in direct sodium–water reaction with dangerous consequences. Hence, by adopting in-bore welding, the tube-to-tubesheet joints are made according to stringent acceptance standards on dimensions and weld quality without any lack of penetration and fusion, cracks, undercuts and unacceptable porosity. The maximum concavity achieved is practically zero and the maximum weld thinning is less than the permissible <0.2 mm. As SGs are made of grade 91 steel, this necessitates a dissimilar joint between stainless steel piping headers of grade 91 steel. The hot-wire NG-TIG welding process was employed for the first time in India for the fabrication of the SG. Figure 6 shows the schematic of the SG as well as tube-to-tubesheet joint details.

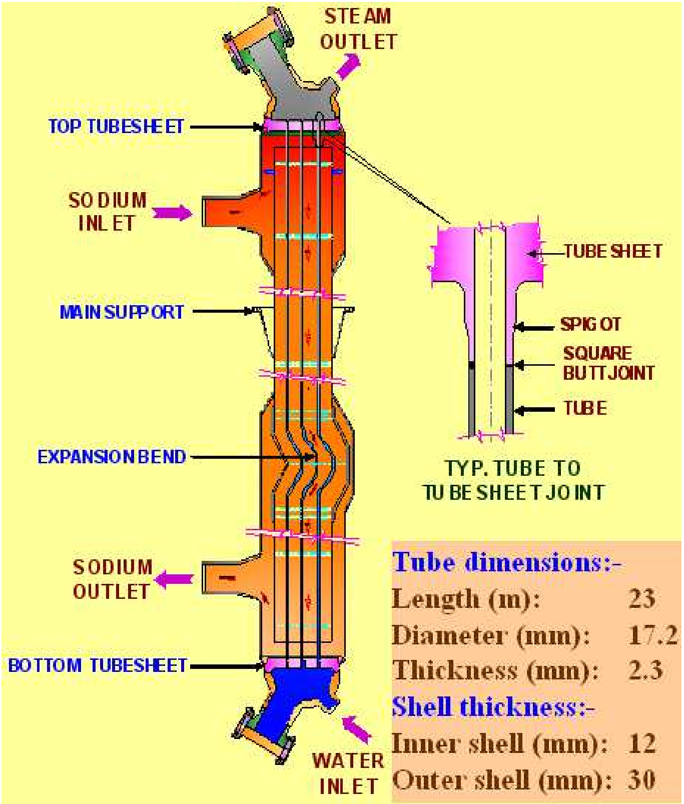


FIG. 6. SG weld joint details.

4.5. Design development and qualification of thermal insulation panels

Austenitic stainless steel plate type insulations are provided in the form of panels around the safety vessel to reduce heat transferred to the reactor vault. As these types of insulation are not commercially available, the same has been designed and fabricated indigenously using 0.1 mm thick sheets stacked to form panels. Dimples are provided to ensure spacing between the sheets. The manufacture and assembly of the panels were completed, overcoming several challenges in view of thin sheet forming, requirement of uniform emissivity, uniform spacing, formation of dimples without cracks, and complicated assembly sequences over the safety vessel. Innovative experiments were carried out to confirm compliance of thermal and seismic design requirements.

5. ERECTION CHALLENGES

In order to complete the activities in time, it was required to carry out civil construction and equipment erection in parallel, involving state-of-the-art erection equipment and construction methodologies and highly optimized construction sequences. The installation of individual permanent components has been completed, precisely meeting the specified erection tolerances towards meeting the functional requirements. Elegant methodology has been finalized for the subsequent erection of the main vessel along with internals and top shield, respecting various erection tolerances and giving due consideration to both time and economy. Transportation of thin shell structures from the site assembly to the support locations is another challenging activity in the construction, executed in an innovative or novel way to achieve economy without affecting safety. Erection of very large dimensioned and slender PFBR components with very stringent dimensional accuracies (a typical tolerance achieved on horizontality over 15 m is less than ±1 mm) is the most challenging task completed for the first time in the country with systematically planned mock-up trials. Computer simulations were extensively used based on 3-D virtual models for establishing erection sequences. Figure 7 shows the erection sequence established for sequential erection of reactor assembly components. Based on these challenges, a document has been made to give guidelines for industries and erection agencies.

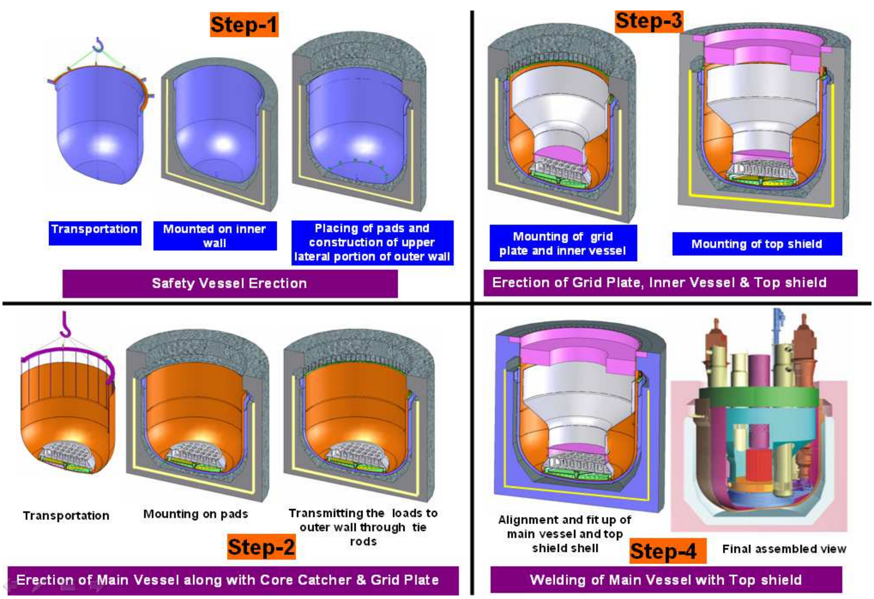


FIG. 7. *Computer simulation of erection sequence of reactor assembly components.*

Handling scheme structures were designed, developed and tested in novel ways to achieve minimum material consumption, no weld attachments and minimum assembly time. Many useful mock-up trials ideal for the training of crane operations were carried out and confidence has been built-up before going for safety vessel handling. It is worthwhile to mention here two typical full scale mock-ups built and employed at IGCAR for the successful erection of the safety vessel (Fig. 8) and welding of the roof slab hanging shell and main vessel (Fig. 9). Furthermore, a full scale mock-up has been built for visualizing the complicated layout of the top shield components (Fig. 10). This mock-up will further help to ensure the availability of space and access for adding and removing complementary shield blocks at any time and smooth operation of trailing cables without any entanglement during rotation of plugs. Computer simulations and mock-up trials helped to ensure good access for critical welds, to establish techniques and tools for the mismatch correction procedure and methodology, and the appropriate welding sequence to minimize distortion controls.

All the above activities are instrumental for the successful erection of reactor assembly components, i.e. the safety vessel, main vessel, thermal baffles, grid plate, inner vessel and roof slab (Fig. 11).

6. IMPORTANT LESSONS LEARNED

6.1. Manufacture of components

Technology development prior to start of construction is essential for long delivery components. The important outcome of the technology development exercise undertaken for PFBR components is illustrated comprehensively in Fig. 12. Judicious choice of tolerances, and the number and location of welds and inspections has to be made. Robust criteria need to be applied for the acceptance of manufacturing deviations and material compositions. Indigenous materials should be used after qualifications of the manufacturing process of direct relevance apart from routine standards. Manufacturing drawings should be finalized after several rounds of discussions with prospective industries, giving due consideration to economy. Subsequently, the revision of manufacturing drawings should be minimized in particular for materials and tolerances. A single industry for the manufacture of permanent components of reactor assembly will certainly help to minimize the integration problems and also to avoid delays in the project. In the case that industry needs technical support, decisions should be taken quickly based on scientific input with due consideration given to international experience.

FIG. 8. Safety vessel erection mock-up.

FIG. 9. Mock-up for welding of roof slab hanging shell and main vessel.

FIG. 10. Full scale top shield layout mock-up.

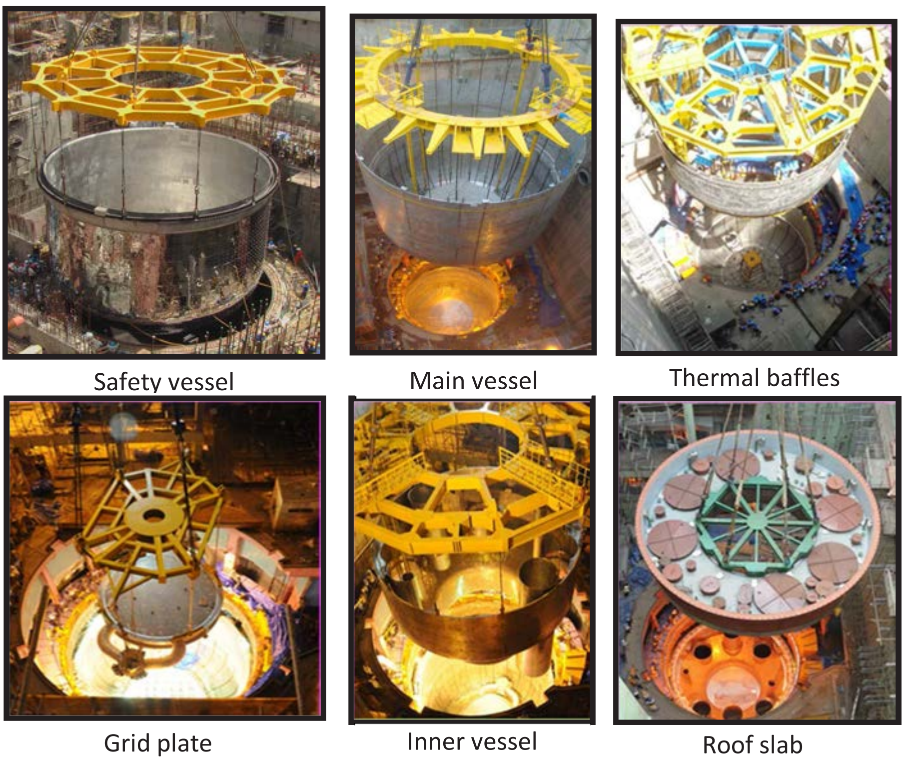


FIG. 11. Erection of reactor assembly components.

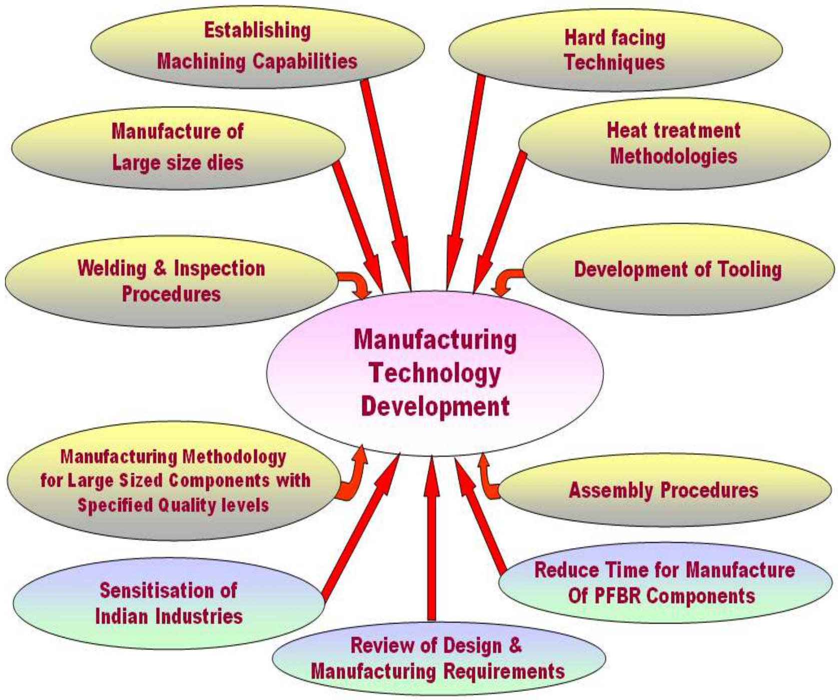


FIG. 12. Outcome of technology development exercise.

6.2. Erection of reactor assembly components

Erection sequences and handling systems should be finalized after detailed discussions with the use of advanced computer software techniques. Care should be taken so that there is no revision of erection sequences and handling systems and handling schemes. Judicious choice of construction sequences of civil, mechanical and electrical systems is essential for success. There is a need for an optimum number of site assembly shops. The possibility should be studied of manufacturing the entire reactor assembly components as a factory made single package item. One possible strategy is schematically explained in Fig.13. The reactor assembly components and civil construction of the reactor vault along with the safety vessel are constructed in parallel in a matching time schedule so that the reactor assembly will be erected without time delay. Subsequently, other reactor internals kept ready in the site assembly shop will be introduced. It is preferred to manufacture the permanent components of the reactor assembly in an integrated manner as a single package involving one major nodal industry.

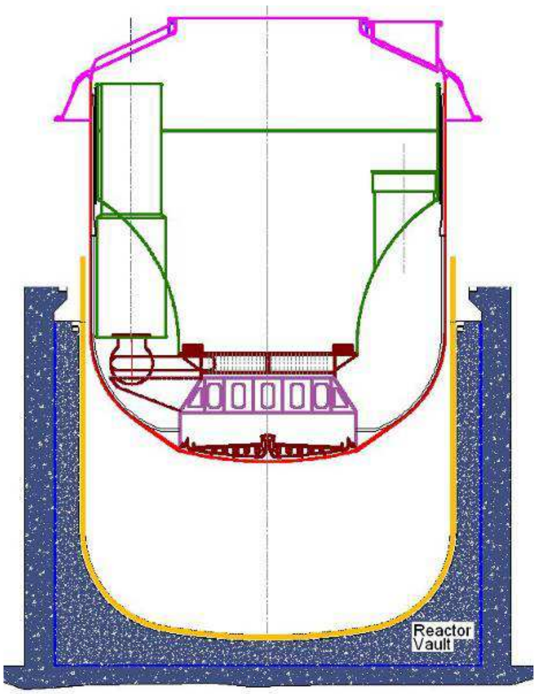


FIG. 13. A proposal for erection of reactor assembly.

Towards achieving this, it is essential to motivate the prospective industries to participate as a consortium attracting good business opportunities in the long term.

7. CONCLUSION

Manufacture of large size thin walled vessels, made of stainless steel with tight form tolerances, machining and assembly of grid plates and SGs with close tolerances are some of the challenging issues which have been successfully resolved for the 500 MW(e) PFBR. In order to build unique capability in Indian industries and also to assess manufacturing tolerances that can be achieved by the industries, elaborate manufacturing technology development works were undertaken prior to starting construction of the PFBR. The welding and pre-service inspection techniques and strategies were finalized through many successful mock-up trials. Judicious choice of tolerances, and number and location of welds and inspections have been made. Robust criteria have been applied for the acceptance of manufacturing deviations and material compositions. Indigenous materials were used after qualifications of the manufacturing process of direct relevance apart from routine standards. From the rich experience gained through the manufacture and erection of reactor assembly components of the PFBR, important guidelines and approaches were derived. The history of design, manufacturing and erection experiences is well documented to adopt for the future application towards achieving economic competitiveness of the technology, respecting the enhanced safety requirements. These are very useful for the success of future nuclear reactors planned by the Department of Atomic Energy.

ACKNOWLEDGEMENTS

The benefits derived from friends and colleagues of the Department of Atomic Energy as well as collaborating individuals, industries and organizations from India and abroad are sincerely acknowledged.

CURRENT TRENDS FOR SODIUM FAST REACTOR DESIGN OPTIONS: AN INDUSTRIAL PERSPECTIVE

J.-P. SERPANTIE, J.-C. LEFEVRE, J.-M. HAMY
AREVA NP,
Lyons, France

Abstract

The fast reactor capabilities of fuel breeding and minor actinide burning give a strong incentive to pursue the development of such reactors in order to keep up with the evolution of the general regulatory environment of nuclear energy and technological progress. Further engineering and R&D efforts have to be invested in order to gain assurance that the benefits expected from the industrial operation of fast reactors will be actually obtained when the need arises. This paper gives an overview of the main current design trends under study in France, and particularly at AREVA, to achieve the high performance level requested by utilities and GENIII+ safety standards. In particular, the issues raised by using liquid sodium as the primary coolant and intermediate heat transfer fluid are presented with regard to safety analysis and response to operational constraints. Examples are given of the rationale for design options that are being assessed by AREVA, from past experience and projects, currently in the framework of R&D actions, to conceptual studies for the ASTRID NSSS and/or more generally for the study of future commercial SFR plants.

1. INTRODUCTION

The history of fast reactors dates back more than 60 years, during which designer teams in North America, Western and Eastern Europe, and Asia have developed various concepts with the aim to finding the best combination of features among many alternatives.

The particular capability of the fast reactor to actually ‘breed’ fissile material, together with generating usable power, was acknowledged already in the very beginning of nuclear energy development. The breeding ratio can be adjusted over a broad range and could even come down to a ‘consumption’ ratio in order to eliminate temporary excessive accumulation of nuclear materials, if so desired. More recently, the capability of the fast reactor to incinerate a large part of the ‘ashes’ coming out of fission reactions in the core, notably the so-called minor actinides (Am and Np), was also acknowledged, thus enabling a significant reduction in the ultimate repository capacity needed to accommodate the remaining radioactive waste.

Both motives give a strong incentive to pursue the development of the fast reactor in order to keep up with the evolution of the general regulatory environment of nuclear energy and technological progress. Further engineering and R&D efforts have to be invested in order to gain assurance that the benefits expected from the industrial operation of fast reactors will be actually obtained when the need arises.

This paper gives an overview of the main current design trends that aim to achieve the desired performance level.

2. SAFETY DRIVEN OPTIONS

2.1. Basic facts

For any nuclear plant, safety has always been a major driver of design decisions. This is especially true for the SFR due to the hard neutron spectrum in the core, and the use of sodium as a coolant, for which no better alternative has been identified thus far.

However, the distinctive features of the sodium cooled fast reactor (SFR) with regard to thermal reactors imply both favourable and unfavourable characteristics.

The combination of core kinetics parameters: effective delayed neutron fraction and neutron lifetime with fast acting reactivity feedback factors, including the Doppler effect for ceramic fuel, makes the transient behaviour of the SFR smooth and predictable, even approaching close to the boundaries of permissible perturbations. In addition, there is no need to cope with the xenon poisoning effect after reactor shutdown. This, together with the large thermal inertia of SFRs means that operators generally acknowledge that fast reactors are easier to control than LWRs.

The liquid sodium also exhibits favourable properties:

- The large margin to boiling temperature: about 300 K above the hottest point in the coolant;
- The low vapour pressure permits operation only slightly above standard atmospheric pressure (just to prevent in-leakage of air);
- The excellent heat conductivity allows a balance to be maintained between power generation in the fuel and the heat removal capability of the primary system over a broad range of normal and out of normal situations;

- Additionally, the large volume thermal expansion factor favours the effectiveness of natural convection cooling in emergency conditions with loss of power supply and fission products are efficiently trapped by the sodium in accidental cases.

At the opposite, as pointed out in Ref. [1], SFRs present some specificities that can make the safety analysis complex and can require more research and advances to make it absolutely robust — which is a must when designing industrial reactors.

2.2. Core safety

The other side of the coin from the 'easy' core controllability is that moving beyond the 'controlled' range can entail severe consequences in terms fuel damage, potentially ending up melting a more or less significant fraction of the core, or even of power excursion in the case of core voiding or core compaction.

This behaviour, where the severity of ensuing events is out of proportion with the magnitude of the deviation from allowable parameters, is commonly termed the 'cliff edge effect'. The first requirement will therefore be to enlarge to the maximum the stability domain, to be sure that normal, incidental and even accidental operating conditions remain far from the 'cliff'.

Then, every initiator and every sequence of events that would result in fuel melting or power excursion must be investigated up to the threshold of technical imagination, in order to demonstrate that the likelihood of severe accidents is extremely small.

In order to comply with defence-in-depth principle, means for severe accident mitigation have to be provided in order to prevent any significant or rapid radiological releases at the site boundary.

Therefore, the strategy must rely on a double-sided approach, where the preventative safety systems are supplemented by mitigating provisions, which are able to cope with a postulated severe accident not related to any credible sequence of events.

This double-sided approach was already introduced on the SUPERPHENIX design, with both preventive (additional absorber devices) and mitigating measures (core catcher, containment resistant to mechanical energy release). It was then pursued and improved on the EFR design, with a 3rd passive shutdown level, a core design aimed to limit the energy release potential at low values in case of CDA, and a resistant primary containment [2]. This approach is pursued with the studies realized in the framework of ASTRID [3, 4] and of future industrial SFRs, which will consider following innovative features as candidates to improve both prevention and mitigation of core accidents:

- On the preventative side, cores with a reduced volumetric power density and a sodium layer above the fuel column are currently proposed in France [5, 6] in order to obtain a low reactivity effect in the case of coolant voiding and generally improved natural behaviour. The main benefit of these provisions is to offer an extended grace period before reaching coolant boiling, thus permitting more efficient action of safety systems, or even operator intervention (e.g. by triggering manual core shutdown). Also, this allows decreasing the potential consequences of abnormal control rod withdrawal. One drawback of these concepts is that the overall size of the core is increased owing to the volumetric power density and to the core height limitation governed by the void effect reduction objective, which induces an increase in the reactor dimensions.
- Another track investigated that could be used to complement or even as an alternative to the preceding one if not sufficient for some initiators or transients, is to add a third shutdown level [7], which would be based on totally diversified design and actuating principles. This system would be actuated directly through physical core parameters (temperature, flow) and independently from the signals delivered by the core instrumentation and trip logic, and be in normal operation already fully inserted in the core to exclude any risk of jamming due to misalignment between core and upper structures.
- On the mitigation side, SFRs are today equipped with retention devices capable of catching up some molten fuel and core structure debris. Such devices have been introduced in earlier SFRs (e.g. Superphénix, SNR 300) and are considered for JSFR, PFBR and BN-800. Core catchers able to retain a whole molten core are now implemented for modern GENIII+ LWRs, such as EPR or ATMEA1. However, robust demonstration of performance is made much more difficult on SFRs than on LWRs by the need to manage a corium with a high Pu content, greater residual power, and presence of sodium. Different possibilities for such a 'robust' core catcher are being contemplated for ASTRID [8] and for future SFRs; installing such a device inside the main vessel or in the reactor pit beneath the vessel bottom. Obviously, the presence of sodium complicates the design of the external catcher and cooling efficiency is a challenge, leading to great complexity, as experienced for the external core catcher of SNR300. This option has to be studied for the case where the internal alternative would not allow meeting the requirements.

- As a conclusion on this point, it is still hard to forecast what will be the optimal design combining core accident ultimate prevention, robust mitigation capabilities and economic efficiency for commercial SFRs. R&D and studies realized in the framework of the ASTRID pre-conceptual design show promise and have now to be confirmed and optimized, with a view to application to commercial SFRs.

2.3. Sodium hazard management

The use of sodium as coolant is necessary for core cooling (primary side) mainly for neutronic reasons. It also has excellent low pressure coolant characteristics, eliminating coolant flashing, and with high thermal conductivity, enabling efficient heat removal from a high power density fast reactor core by single-phase coolant. With proper control of oxygen impurities, it has excellent compatibility with austenitic and ferritic cladding and structural materials.

However, difficulties linked to liquid sodium use are well known, owing to its chemical reactivity with oxygen-containing compounds, mainly air and water.

Since the purpose of an SFR power plant is to generate electricity from a turbine plant operated classically through a water–steam Rankine cycle, the risk of sodium–water reaction cannot be excluded in the event of steam generator failure, inasmuch as sodium is also utilized as the heat transfer fluid in the intermediate system between the primary system and the water–steam plant. All past and current SFR power plants use an intermediate sodium system. This option has been validated through quite good experience feedback, with a limited number of sodium–water reactions on operating SFRs efficiently managed by the ad hoc safety provisions.

The same holds true regarding secondary sodium fire hazards, considering that primary sodium fires are excluded by selecting the integrated primary system (‘pool’) concept, or filling all rooms where primary sodium could leak from with inert gas and considering that secondary sodium fires are only chemical or thermal loading events, without radiological consequences.

However, the context now is evolving towards more stringent regulations and application rules regarding the release of soda based chemicals in the environment [1]. In addition, strong improvements on the handling of sodium–water reaction and sodium leaks are judged necessary, in relation to public and operational acceptance, and the safety significance of sodium related hazards must be re-evaluated.

This leads to imposing more severe requirements on the design of the intermediate heat transport system, resulting in the following features studied, as an example, for ASTRID:

- Sodium–water reaction prevention: A drastic provision would be to eliminate the sodium in steam generators. During the R&D phase, no other liquid metal has been identified which could provide decisive advantages in replacing sodium [9]. The use of gas (nitrogen) instead of water as tertiary working fluid is still currently being assessed for ASTRID [10]. This eliminates most concerns on this aspect, but raises new technical issues such as the need to develop a closed loop BRAYTON cycle system adapted to industrial power ranges, compact sodium–gas heat exchangers, management of high pressure gas in connection with sodium, etc. It is in particular estimated that direct heat exchangers between the primary system and pressurized gas loops would not be acceptable, thus keeping the need for an additional intermediate heat transport system in sodium. An evident economic penalty is the reduced thermal efficiency of this cycle (~37% instead of ~42%). Pursuing studies is therefore necessary to assess the economic penalties before being able to select the best concept for industrial applications.
- Sodium–water reaction mitigation: For the case where the classical steam water cycle is selected, the AREVA current proposal [11] is to provide modular steam generators, in order to add an inherent line of defence in addition to the prevention and control lines of defence. The size of each module (typically 150–200 MW(th)) is such that the largest possible sodium–water reaction, corresponding to a failure of the whole tube bundle, can be accommodated without excessive loadings on the steam generator casings, the secondary sodium piping system, and the intermediate heat exchangers within the primary system. This avoids the need for a difficult demonstration on the limit of propagation of failures from one tube to the neighbouring tubes in a large bundle. Modular steam generators have been implemented on PHENIX and on the Russian BN-600 reactor, and proved very effective in achieving good overall plant availability in spite of experiencing some sodium–water reactions, which were overcome. This option, although more expensive than the monolithic steam generators of SUPERPHENIX (750 MW(th)) or EFR (600 MW(th)) have some operational advantages such as easier maintenance or replacement or requalification. They can also allow reducing the height of buildings housing the secondary sodium system and to operate at part load in case of difficulty on a steam generator.

Another possible design that has been evaluated is the double-walled tubes in the steam generator to reduce the likelihood of failure, bringing direct contact between sodium and water. This was implemented in the EBR II reactor in the USA, which did not show any sodium–water reaction over its whole lifetime. However, the demonstrative value of EBR II experience is questionable as the operating parameters of this reactor were ‘milder’ compared to other SFRs (sodium temperature at steam generator inlet: 467°C as opposed to 550°C on PHENIX), thus putting less strain on the equipment. In addition, it was judged by the French partners during the R&D phase that the double-walled tube effectively reduces the likelihood of sodium–water reaction, but not to the extent that this event could be considered as practically eliminated. Its much greater cost, together with anticipated strong manufacturing difficulties and other issues such as the evolution of inner–outer tube bonding during service life, the possible degradation of heat exchange performance, the difficulties for in-service inspection of both tubes and early detection of an incipient failure of the inner tube before propagating to the outer tube did not lead AREVA to propose this option for ASTRID and for future commercial SFRs.

The concept of an ‘inverted steam generator’ is also evaluated by AREVA in its own R&D programme and as an alternative option for ASTRID or future SFRs [11]. Contrary to common practice, this concept is based on the idea of placing the sodium inside the heat exchange tubes, and the water–steam outside the tube bundle, enclosed by the steam generator casing. A realization of it has been implemented in the Russian BOR-60 test reactor. Basically, this concept is intended to intrinsically limit the steam generator failure to a single tube, since the high pressure on the water-steam side should prevent the sodium from getting into the casing. An obvious disadvantage is the need for a stronger steam generator casing capable of accommodating the full pressure of the water–steam system plus any additional loadings from the sodium–water reaction. A possible outcome of the assessment might be that the unit size of steam generator is limited due to manufacturing difficulties (especially for the tube plates). Should this size be too small, the cost benefit analysis would not be improved compared with the ‘normal’ modular steam generator design of 150/200 MW(th) unit size.

- Sodium fire prevention: In this respect, sodium also offers favourable properties as sensitive leak detection systems can be implemented, based on exploiting its electrical conductivity which shorts detecting wires wrapped around piping or detecting plugs at the bottom of vessels, and the opacity of released smoke which is detected by light monitoring in the premises. An effective prevention means can be to provide guard pipes/

vessel with inert gas filling of the inter-space round the secondary sodium system, as is done for the primary sodium system. This feature is proposed for the Japanese JSFR reactor. It is being assessed in studies for ASTRID and for future commercial SFRs in order to examine the balance between the increased complexity of the design and erection work of the sodium systems and their fastening to building structures, the increased difficulty regarding accessibility for in-service inspection, and the benefits in terms of eliminating some sodium fire related loading cases on the premises, thus facilitating the safety analysis as a whole.

- Sodium fire mitigation: The best response when detecting a sodium leak or sodium fire start is to dump the sodium system as quickly as possible, and to re-configure the ventilation system in such a way as to limit the input of air and the dispersion of chemically aggressive products in the form of aerosols to other premises. The sodium containing rooms must also have limited air content (i.e. limited volume) in order to limit the sodium fire duration simply by smothering. This philosophy was implemented on SUPERPHENIX [12] and is pursued for the NSSS ASTRID design, with improvements such as reducing as much as possible the length of sodium piping, installing them within small-sized galleries to limit the feeding of air to the sodium fire, or filling these galleries with inert gas if applicable, considering the need for providing a leaktight liner on to concrete walls together with leaktight penetrations at piping inlet/outlet. In addition, the strong reinforcement of environmental regulation mentioned above will lead to the proposal of additional means to collect and limit sodium releases that could result.

Besides reactor shutdown, the decay heat removal (DHR) and containment of radioactive nuclei are the other main safety functions.

2.4. Decay heat removal

As stated in Ref. [1] DHR capability loss must be practically eliminated on SFRs. This leads to the same trend as for the core shutdown function toward improving reliability by basically providing two redundant and diversified cooling systems removing heat directly from the primary system and additional diversified solutions for long term operation. For the EFR project [2], this was, for example, ensured by two fully diversified systems, each consistent with the single failure criterion (i.e. 3 × 50%), one operating in natural convection, the other in forced convection (but capable of natural convection in degraded or long term mode) and additional diversified possibilities through the steam generators or reactor pit cooling.

For ASTRID and future commercial SFR plants, the same approach will be followed, with the will to still reinforce the diversity between systems and the consideration of potential common cause failures. This can, for example, protect and separate the DHR piping above the reactor roof, and diversify and protect sodium–air heat exchangers against external threats, etc.

The reliability of the DHR function also depends on maintaining the primary sodium level at a sufficient elevation to prevent uncovering of the fuel subassemblies (at least the fissile column) and of the DHR heat exchangers. This is achieved in the first instance by a guard vessel situated at a close distance to the main reactor vessel. For extremely unlikely events, such as the leakage of both the reactor and guard vessels due, for example, to a delayed or undetected defect on the guard vessel, the reactor pit arrangement must be designed in such a way as to provide sufficient minimal covering of core and DHR system. This requirement was already taken into account for the EFR design and is judged necessary to be taken into account for ASTRID and for commercial SFRs. The EFR 'anchored safety vessel' design [2], although quite economic, is, however, questioned, as design robustness is difficult to demonstrate, and in-service inspection quite impossible. This can result in different designs, for example, a classical safety vessel and a 'narrow' reactor pit with a simple metallic liner to cope with multiple leak issues.

2.5. Containment

The containment is a tricky issue for the SFR, since it is hardly conceivable for an industrial size SFR to put fast acting isolation valves on the main piping of the intermediate sodium system at the crossing of the containment boundary. The technology of such valves that exist on the secondary water–steam system of LWRs is not transferable to the sodium system. On the other hand, the accident analyses show that there is no situation causing a high pressure rise within the whole reactor building of the SFR as it occurs on the LWR, owing to all provisions presented above, which aim to limit the loadings generated by a failure in a sodium system to the closest possible to fault location, and by managing hydrogen risk. In this respect, the pool-type SFR concept that is retained for ASTRID, in line with most previous projects in Europe and the Russian Federation, offers an advantage by concentrating within a limited plant area all of the primary sodium, which is a potential source of radioactive product release.

Safety analyses also consider externally induced hazards, such as aircraft crash, earthquake, flooding, etc.

The most sensitive areas of the nuclear plant are to be protected against an aircraft crash by a strong enclosure providing mechanical resistance to the impact forces.

The fact that sodium systems do not require pressurization, which is basically an advantage for the mechanical design, could be seen as a disadvantage from the viewpoint of earthquake resistance, as the earthquake-induced loadings appear more severe in relation to normal operational loadings, contrary to the case for LWRs where pressure loadings dominate by far. A workaround to this issue is to provide seismic isolating pads underneath the basemat of a nuclear island, which in addition allows improved robustness, in particular in the frame of the post-Fukushima analysis.

Concerning the protection against external flooding hazards, the same principles are applicable to SFRs as to any other nuclear plant, namely, setting the general site level at a sufficient elevation and providing additional embanking or leaktight and bunkerized buildings with protected openings (c.f. EPR), with, of course, particular attention paid to sodium systems (storage tanks, etc.).

The aftermath of naturally occurring accidents could be the loss of station service power and loss of the main heat sink due to infrastructure destructions over a large area surrounding the plant. In this respect, the natural convection capability of sodium systems is a distinct advantage for the SFR by reducing the emergency source requirement compared to other nuclear plants. However, this capability can be maintained even in extreme conditions (extreme cold winter, snow storm, aircraft crash, etc.). This will require adequate autonomous control systems to allow the DHR systems to operate without additional power source and protection and diversity of the air inlet/outlet openings.

3. OPERATION DRIVEN OPTIONS

Thus far, this review has covered mostly safety related topics, according to their utmost significance for the design. However, strong improvements are also expected for SFR industrial plants in the operational performance domain and in the capability of in-service inspection and repairs [13].

As could be expected, the areas that give rise to particular difficulties and where more efforts should be deployed are those impacted by the use of sodium coolant. Some examples are given below.

3.1. Fuel handling and storage

The spent fuel subassemblies discharged from an SFR core must reside in interim storage under sodium, until their residual power has become low enough to enable their washing and subsequent storage in a water pool on-site, prior to shipment to the reprocessing plant.

Interim storage in sodium can be provided within the reactor vessel itself. Depending on the fuel management strategy, the number of subassemblies in one reloading batch is usually sufficiently low to be accommodated in positions around the core, external to the lateral neutron shielding elements. Obviously, this appears the most attractive solution, from the viewpoint of investment cost reduction. This cost efficient option was firstly adopted for SUPERPHÉNIX as replacement for the failed in-sodium storage drum, then adopted on the EFR project [2] and was selected for the PFBR and the BN-1200 designs. The JSFR design keeps an ex-vessel in in-sodium storage.

The main objection to the in-reactor spent fuel storage is that it offers no flexibility to cope with any fault which would jeopardize the availability of the fuel transfer systems to washing pits, or of the washing pits proper. The fuel reloading operation must be stopped until the fault is cleared, which could penalize the availability of the power station.

More importantly, a new requirement has emerged from utilities imposing that the discharge of all core subassemblies should be possible within a limited time scale, to cope with unforeseen events or faults causing the need for in-reactor intervention for extensive inspection and/or repair. External in-sodium storage can facilitate or even be needed to meet such a requirement, particularly if high residual power subassemblies had to be handled, as is the case, for example, of fuel assemblies containing a significant percentage of minor actinides (~10%).

For ASTRID, the external storage option is currently preferred due to this last requirement and to the provision of minor actinide-bearing subassemblies. It will also contribute to the fuel management flexibility, desirable for an experimental reactor that has to easily load/unload experimental subassemblies or irradiation samples.

However, this option will incur a significant cost, the conception of the storage having to include safety features equivalent to the reactor safety features, in particular for DHR, maintaining coolant inventory and protecting against external or internal events. It is probable that for future commercial reactors the in-vessel storage option will be preferred, in particular if this avoids managing fuel subassemblies with high actinide content, the requirement of rapid total core unloading being managed through particular provisions such as specific sodium filled casks, as described in Ref. [14].

3.2. Component handling

The main components of the SFR primary system (pumps, intermediate heat exchangers) are slender objects characterized by a large height-to-diameter ratio. When the need arises for removing one of these for maintenance, the usual practice is to handle it vertically, enclosed in a leaktight and radiation shielded flask, up to washing and decontamination pits, and maintenance shop on-site.

In the framework of optimization studies of the nuclear layout, the relative merits of locating the component maintenance facilities inside or outside the nuclear island building are being evaluated. The latter solution has advantages in reducing the room requirement within the nuclear island area, and offering the possibility for sharing these facilities between several SFR plants at the same site.

The issue to be resolved then is designing the penetrations through the containment boundary and the aircraft crash barrier, considering that creating openings with a large height-to-width ratio is an issue for the mechanical design of building structures, especially regarding the A/C crash resistance.

An alternative option being investigated is handling the components in a horizontal position and carrying them through a cylindrical maintenance lock, as is done on the LWR. However, there again, the SFR takes its toll by requiring special provisions for collecting dripping primary sodium coolant and the horizontal support of mechanically ‘weak’ components.

3.3. In-service inspection and repair

In-service inspection and repair has always been and remains a difficult issue for the SFR, especially concerning the reactor internal structures. The current technology relies mostly on the use of ultrasonic devices, the performance of which are being continuously improved thanks to R&D efforts on the sensors and the electronics for signal treatment. Remarkable achievement have been demonstrated on PHENIX by the successful inspection of the core support structure, continuous monitoring of subassembly position, etc. These capabilities are currently being extended, through the development by AREVA and CEA of US sensors able to detect flaws in the sodium environment [15].

In order to take full benefit of the capabilities of US inspection devices, the design of future SFRs must provide extensive access to internal structures, notably down to the core support and internal core debris catcher to ensure their keeping in good condition. When taken at the early stage of the design, these requirements lead to favourable arrangements of the internals within the primary vessel with, for instance, a specific alignment between the access locations on the roof slab and the sensitive zones to be controlled. ASTRID will have to demonstrate such in-service inspection and repair capability.

4. CONCLUSIONS AND PERSPECTIVES

At the end of this overview of the main features of SFR technology, one could gain the impression that the use of sodium coolant is the source of a number of difficulties, and that other coolants should be preferred. Actually, no better alternative could be found after more than 60 years of fast reactor development, starting with EBR I in the USA (which, incidentally used NaK coolant). It could be said that this is the price paid for benefiting from the unique capabilities of the fast reactor concerning the management of nuclear material resource, on the one hand by saving on consumption and avoiding exhaustion thanks to breeding, and on the other hand by reducing the ultimate radioactive waste thanks to multi-recycled plutonium consumption and eventually minor actinide incineration.

The realizations and projects that have existed over this long history show that SFR feasibility is not an issue. What is at stake now is demonstrating that a fast reactor design close to industrial criteria can be developed, compliant with the latest safety requirements (those of Gen III+) and meeting the utilities' expectations regarding operability, performance and cost efficiency. The continuing studies and future construction of ASTRID will materialize an essential further step on the way of progress.

Taking into account safety requirements of ever increasing severity, in particular for specific provisions needed by the sodium coolant and by the fast neutrons core characteristics, it can be anticipated that the resulting SFR industrial designs will utilize more commodities (steels and other metals, concrete) per unit MW(e) installed, compared to the Gen III+ LWR, with often also more expensive technologies (stainless steel, heated piping, etc.). Therefore, it can hardly be expected that the investment costs of both technologies will come out at par.

But the economic merit should not be judged only from the viewpoint of the MW·hour produced, but also by taking into account the savings realized on natural uranium procurement and enrichment; recycled plutonium storage, and minor actinide storage. The vision is that the SFR will have an industrial future in countries committed to nuclear energy in the long term, alongside with Gen III+ reactors. The time frame and deployment rate will depend on the operational records of the new generation SFR prototypes in the world (PFBR, CEFR, BN-800, ASTRID) and on the licensing, social acceptability and economic projections that will be extrapolated from these experiences — which have therefore to be success stories. In such a case, we can imagine a deployment in two or three steps, to be analysed and refined through detailed scenario analysis, including economic evaluations:

- A first deployment step that could have as its main objective provision of flexibility to the nuclear cycle management, and in particular to multiple recycling of the plutonium resulting from LWR MOX used fuel, which would motivate in France, for example, construction of a small number of SFRs, complementary to Gen III+ or advanced LWRs loaded with UOX and/or MOX. At this stage, the breeding gain could be optional and adjusted to the isotopic quality requested for the global Pu inventory.
- A second deployment step, over the longer term, that would have as a main objective the limiting, significantly, of the need for natural uranium supply to prevent the foreseeable shortage of uranium, to increase energetic independence, to stabilize the plutonium inventory, and eventually to limit the minor actinides inventory to be stored in a geological deposit. For this second step, a significant part of the nuclear fleet (>30%) would have to be converted into SFRs, advanced LWRs loaded with UOX and MOX continuing to operate in symbiosis with SFRs. At this stage, a breeding gain becomes valuable, to allow improving the global isotopic quality of Pu stocks (degraded by LWR MOX or to prepare an eventual step 3). This second step relies on a composite reactor fleet (LWR + SFR), thus allowing the cost penalty due to SFR operation to be limited.
- A third step with only SFRs remains ultimately possible in the case where global economic and operational performance of both LWR and SFR systems will come at par resulting on the one side from natural uranium shortage and very high associated prices of the fissile resources, and on the other side from simplifications, advances and experience gained through the previous steps. In this case, the SFR should become the preferred technology, due to its unique capability to breed fissile resources and to limit minor actinide production, or even reduce their inventory.

Then, it can be said that the extra costs of SFR demonstration and prototype plants to be built over the short to mid-term can be seen as the price of an extendible ‘real option’ that reflects the value of maintaining open the possibility to later invest more massively in commercial models that will have taken benefit from continued improvements to allow sustainability of CO_2-free electricity production for future centuries.

ACKNOWLEDGEMENTS

The authors thank the contributors to the papers cited in the reference list, essentially from CEA, EDF, AREVA and IRSN, for their contributions to the progress on SFR development and assessment.

REFERENCES

[1] FRENCH INSTITUTE FOR RADIOLOGICAL PROTECTION AND NUCLEAR SAFETY, Overview of Generation IV (GENIV) Reactor Designs — Safety and Radiological Protection Considerations, Rep. 2012/158, Fontenay-aux-Roses, France (2012).

[2] LEFEVRE, J.C., European Fast Reactor: Outcome of Design Studies, EFR Associates (1998).

[3] ALPHONSE P., et al., "Status of ASTRID Architecture and pre-conceptual design", paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[4] SAEZ M., et al., "Status of ASTRID nuclear island pre-conceptual design", paper presented at. Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[5] CHENAUD M.-S., et al., "Status of ASTRID core studies at the end of predesign phase 1", paper presented at. Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[6] VERRIER D., et al., "A New Design Option for Achieving Zero Void Effect in Large SFR Cores", paper presented at. Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[7] BEILS S., et al., "Safety objectives of SEPIA system for Sodium Cooled Fast Reactor", Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP), San Diego, CA, 2010, American Nuclear Society, LaGrange Park, IL (2010).

[8] SERRE F., et al., "R&D and Design Studies for the ASTRID Core-Catcher", paper presented at. Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[9] BRISSONEAU L., et al., "The potential use of an alternative fluid for SFR intermediate loops: selection and first designs", Proc. Int. Conf. on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), Kyoto, Japan, 2009, IAEA, Vienna (2012).

[10] LAFOND G. et al., "ASTRID power conversion system: assessment on steam and gas options", paper presented at. Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[11] GERBER A., et al., "Safety improvement research to design a SFR Steam Generator with regard to sodium/water reaction risk", Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP), San Diego, CA, 2010, American Nuclear Society, LaGrange Park, IL (2010).

[12] NATTA M., "Safety Issues for LMBFR: Important Features Drawn from the Assessments of Superphènix", Proc. Workshop on Advanced Nuclear Reactor Safety Issues and Research Needs, Paris, 2002, OECD Publishing, Paris (2002).

[13] SAUVAGE J.-F., et al., these Proceedings.

[14] CHASSIGNET M. et al., "Challenges and Innovative technologies on fuel handling systems for future Sodium cooled Fast Reactors", Proc. Int. Conf. on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), Kyoto, Japan, 2009, IAEA, Vienna (2012).

[15] SIBILO J. et al., "Generation IV nuclear reactors: Strategy and challenges of R&D program for improving Inspection and repair of sodium cooled systems", Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP), San Diego, CA, 2010, American Nuclear Society, LaGrange Park, IL (2010).

OVERVIEW OF THE US FAST REACTOR TECHNOLOGY PROGRAMME

R.N. HILL
Argonne National Laboratory,
Argonne, United States of America

Abstract

This paper provides an overview of fast reactor research and development (R&D) efforts in the United States of America. The diverse R&D activities are funded by several programmes in the US Department of Energy's nuclear energy portfolio. Innovative technology options that may yield significant cost reduction benefits have been identified through concept development studies: high strength structural materials, a supercritical CO_2 Brayton energy conversion cycle, and advanced modelling and simulation techniques. In addition, technology development efforts for safety and licensing, improved recycle fuels, and under-sodium viewing are ongoing. For each area, recent accomplishments and key facilities are identified to provide an indication of current status.

1. INTRODUCTION

Fast reactor research and development (R&D) is motivated by the fundamental physics characteristics (e.g. significant excess neutrons, high fission ratio, high neutron leakage) that provide favourable performance for diverse energy production and fuel cycle applications. Fast reactors are envisioned for a wide variety of actinide management strategies ranging from actinide destruction in closed fuel cycles to enhanced uranium utilization. The fast spectrum physics can also be exploited to enhance inherent safety behaviour and/or extend the cycle length and burnup; such features are particularly important for small reactor applications. With successful technology development, fast reactors are also intended for electricity and heat production missions, as being pursued in the Generation-IV International Forum collaborations.

A variety of fast reactor concepts with different technology options (e.g. coolant) and diverse missions (e.g. once-through high burnup, small modular) are being proposed by US industry. Most of the United States of America's fast reactor R&D activities are being conducted as part of the US Department of Energy's Office of Nuclear Energy (DOE-NE) Advanced Reactor Concepts (ARC) Program. The mission of ARC is to develop and refine future reactor concepts that could dramatically improve nuclear energy performance (e.g. sustainability, economics, safety, proliferation resistance). Several important fast reactor technology issues are being pursued in other DOE-NE programmes: the supercritical CO_2 Bratyon cycle in the Advanced Small Modular Reactor (SMR) Program, recycle fuels in the Fuel Cycle Technology (FCT) Program, and advanced reactor modelling techniques in the Nuclear Energy Advanced Modeling and Simulation (NEAMS) Program. In this paper, the current status of the complete suite of US fast reactor work is summarized.

Because capital investment in reactors is the dominant cost of any nuclear fuel cycle, R&D efforts to improve system performance are the primary focus. Demonstration sodium cooled fast reactors (SFRs) have been successfully built and operated in worldwide fast reactor programmes, establishing the feasibility of base SFR technology. A variety of innovative technologies that hold the promise for significant cost reduction are being researched, as described in Section 2.

Other key technology issues that must be addressed for successful application of fast reactor technology include: safety assurance and licensing framework, development of robust recycle fuels, and improved maintenance and inspection techniques. US efforts in these three areas are described in Section 3.

2. TECHNOLOGY INNOVATIONS FOR CAPITAL COST REDUCTION

2.1. Concept development studies

Concept development studies guide fast reactor research by providing a fundamental understanding of the technical utilization and feasibility of advanced technology options. In recent years, a broad range of innovative technologies and refined design features have been conceived including compact fuel handling, vented fuel, twisted tube heat exchangers, electromagnetic pumps, alternative heat transport and containment configurations. These innovations support cost reduction by system compaction, reduced commodities, and/or improved reactor performance. The concept research evaluates the feasibility of incorporating these innovatives features in an integrated reactor system, and estimates both the benefits and design impacts.

Previous concept development studies confirmed the potential benefit of advanced structural materials and advanced energy conversion (technology development efforts in these areas are described in Sections 2.2 and 2.3, respectively). Recent US work has focused on small (~100 MW(e)) reactors with unique features such as long lived cores [1]. Compact configurations with refined fuel handling mechanisms or upper internal structure are being explored. Recent evaluations include the utilization of improved intermediate heat exchangers [2], and conversion of a small SFR to thorium fuel [3].

Studies of the integrated reactor system also point to the importance of technology maturation for new mechanisms and reactor components that promote system compaction. Thus, a new intermediate scale facility for testing in sodium is being developed. The Mechanisms Engineering Test Laboratory (METL) will test small and intermediate scale sodium components, instrumentation, and inspection/repair technologies using a flexible multi-vessel testing station configuration. Efforts on fast reactor knowledge recovery and information preservation are also conducted [4, 5] to prevent loss of fast reactor historical operating and design information and to convert key testing data into modern database structures allowing its use for validation (see Section 3.1 for examples).

2.2. Advanced structural materials

Advanced materials are a critical element in the development of advanced fast reactor technology. Enhanced structural materials can improve performance 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 application.

Over the past few years, advanced ferritic-martensitic and austenitic alloys with improved strength have been developed. First a comprehensive assessment of key material properties and anticipated performance of high strength modern materials in SFR operating environments (e.g. heat exchangers, piping, steam generators) identified a small set of candidate alloys. Scoping tests of tensile, creep, creep fatigue, Charpy impact, thermal ageing, weldability, and sodium compatibility were conducted. Based on these results, a ferritic martensitic steel Grade 92 with optimized composition and thermomechanical treatment, and austenitic stainless steel NF709 have been identified for extension into the performance testing phase [6].

The relevant SFR materials requirements and experience have been reviewed in detail [7]. To take advantage of improved strength and temperature performance, improved techniques for materials qualification are also vital. Significant progress has been made in understanding high temperature damage mechanisms (e.g. creep fatigue), and predicting lifetime of advanced structural alloys under thermomechanical loading conditions relevant to the SFR operating environment [8].

2.3. Supercritical CO_2 Brayton energy conversion cycle

Advanced energy conversion systems such as a supercritical CO_2 Brayton cycle offer the promise of improved thermal efficiency and significantly reduced size and cost. Concept studies clearly indicate that both new turbomachinery and improved heat exchangers are needed to realize major benefits.

A fully recuperated split-flow demonstration of supercritical CO_2 turbomachinery is being conducted at the 1 MW(t) scale [9]. This loop shows the behaviour for operation near the critical point and confirms the thermodynmics; testing of a variety of control and operational techniques has started [10]. Plant dynamics models for coupling to a SFR have been developed and validated. Recent work includes a unique capability to control the entire range from full power to decay heat removal levels [11]. A new test loop for sodium–CO_2 interactions under prototypical SFR and supercritical CO_2 conditions has also recently been completed.

To effectively utilize these secondary systems in fast reactors, advanced heat exchangers and compact components are also being pursued. The small scale sodium plugging test loop has demonstrated that clean sodium (~1 ppm oxygen) does not plug even for very small (2 mm) diameter channels. Plugging tests for a wide variety of operating conditions are being conducted with controllable heat removal capabilities that allow temperature control at the plugging site [12]. Testing apparatus for fundamental testing on related issues such as freeze/thaw and draining/refill behaviour is nearly complete.

2.4. Advanced modelling and simulation

Fast reactor simulation requires modelling diverse, coupled physics, including neutronics, thermal fluid dynamics and structural phenomena. Improved modelling will make reactor design tools more predictive, promoting design optimization and reducing reliance on calibration and conservative margins. Advanced neutronic methods have been developed and validated by comparison to previous fast reactor critical experiments [13]. Modern computational fluid dynamics techniques have been employed for detailed modelling of wire-wrapped fast reactor assemblies allowing a better understanding of the turbulent flow and heat removal behaviour. Advanced structural mechanics codes are being developed to address crticial phenomena such as core deformation leading to reactivity feedback [14]. A fully coupled multi-physics, multi-scale simulation of an entire reactor core will be demonstrated this year by coupling the PROTEUS transport suite, NEK5000 CFD package, and the DIABLO structural dynamics code using the MOAB computational framework.[15]

Improved nuclear data are important for both system optimization and safety assurance. New data are being obtained and evaluated for key actinides and materials that impact important parameters such as criticality, transmutation rates and reactivity feedback coefficients. Recent studies using improved methods and newly evaluated nuclear data covariance files, together with selected integral experiments, indicate uncertainties in predicted values can be reduced to the range of the uncertainties of the integral data sets [16]. In addition, R&D is being conducted to develop cutting edge measurement devices. The scale-up to a half instrumented Time Project Chamber has been successfully completed and the system has been collecting data in the upgraded facility at LANSCE since August 2012 [17].

3. OTHER FAST REACTOR TECHNOLOGY RESEARCH

3.1. Safety and licensing

Inherent safety is a key approach for licensing assurance and robust system performance. Recent IAEA benchmarks on natural circulation tests conducted in the Monju [18] and Phenix [19] fast reactors, and metal alloy fuel pin disruption tests at TREAT provide key validation data. A new IAEA coordinated research project was launched in June 2012 to evaluate the EBR-II shutdown heat removals tests. The detailed test data were recovered and a benchmark created from these 1986 tests, with very severe transients, including an unprotected loss-of-flow from full power [20]. Beyond the IAEA benchmarks, other US experimental facility (TREAT, EBR-II, FFTF) testing data are being recovered in modern database formats.

Another significant effort recently completed was an evaluation of SFR licensing gaps [21]. Five technical topical panels were assembled to evaluate accident sequences, sodium phenomena, computer codes and models, fuels and materials, and source terms. Stewardship of fast reactor knowledge and experimental facilities were identified as key challenges. Based on the identified licensing priorities and funding resources, initial work has started on the knowledge preservation activity, and improvement of US analysis tools for fast reactor transients and regulatory source terms.

3.2. Fuels development

US fast reactor fuels development is covered in detail in a related paper at FR13 [22].

The objective of fast reactor fuels R&D is to develop transmutation fuels for use in fast reactors with associated closed fuel cycle. This fuel must allow 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 is being conducted on metal alloy [23] and

MOX [24] 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 [25]. Advanced fabrication technologies are being developed to allow remote operation, minimize losses and wastes, and streamline operations [26, 27].

Similar to the reactor modelling effort, extensive theory and modelling efforts of fuel behaviour and performance are also being pursued to understand and optimize performance of diverse options [28–30].

3.3. Under-sodium viewing

Because international demonstration and test fast reactors have encountered difficulties related to the inspection and repair of system components, under-sodium viewing is an important technology for SFRs in order to demonstrate their reliability. US research has explored both new transducer designs and ultrasonic waveguide devices. A wide variety of material and configuration options have been assembled and tested. Early work was conducted on single transducer elements to demonstrate viability, with extension to linear array devices. In current work, prototype array detectors are being tested in water and sodium for both demonstration and imaging refinement [31].

4. SUMMARY

The objective of the US fast reactor technology programme is to research and develop advanced technologies to significantly improve economic and safety performance of fast recycle systems. Concept development studies are conducted to evaluate a wide variety of technology options and assess the performance impacts on an integrated reactor system. Based on this work, several innovative cost reduction technologoes are being researched:

- *Advanced materials.* Candidate high strength ferritic and austenitic alloys have been identified for performance testing in the SFR environment.
- *Advanced energy conversion.* Focus is on supercritical CO_2 Brayton cycle with a small scale turbomachinery loop operating and compact heat exchangers being developed.
- *Advanced modelling and simulation.* Improved accuracy of methods and data, and better integration of multi-physics models will promote design optimization.

Other R&D activities address key known technology challenges:

- *Safety and licensing.* The focus is on validation of an inherent safety approach to prevent severe consequences; key licensing gaps have been identified.
- *Fuels development.* Recycle fuel fabrication and irradiation performance is being studied, with a concurrent development of advanced modelling tools.
- *Undersodium viewing.* Modern ultrasonic techniques are being explored for this important maintenance and operations capability.

In conclusion, the US fast reactor contributions are focused on the development of required base technical capabilities and a small set of promising innovative technology options. Because of their flexible capability for actinide management, fast reactors are envisioned for a wide range of fuel cycle missions. Thus, fast reactor demonstration and deployment awaits US fuel cycle strategic and deployment decisions. The lack of a current demonstration reactor plan allows time for the innovative R&D to mature, but hampers practical confirmation of technology options and performance.

ACKNOWLEDGEMENTS

This work was supported by the US Department of Energy, Office of Nuclear Energy, Advanced Reactor Concepts (ARC) and Advanced Small Modular Reactor (SMR) R&D Programs.

REFERENCES

[1] GRANDY, C., KIM, T.K., et al, “Advanced Fast Reactor-100 – Design overview”, paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[2] FARMER, M., GRANDY, C. et al., “Intermediate Heat Exchanger (IHX) trade study in support of Advanced Fast Reactor development”, paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[3] HEIDET, F., KIM, T.K., GRANDY, C., “Feasibility study on AFR-100 fuel conversion from uranium-based fuel to thorium-based fuel”, paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[4] OMBERG, R et al., these Proceedings.

[5] SACKETT, J et al., these Proceedings.

[6] SHAM, T.-L., TAN, L., YAMAMOTO, Y., “Development of advanced 9Cr ferritic martensitic steels and austentici stainless steels for sodium-cooled Fast Reactors”, paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[7] NATESAN, K., and Li, M., these Proceedings.

[8] LI, M., MAJUMDAR, S., NATESAN, K., Modeling Creep-Fatigue Behavior in Mod.9Cr-1Mo, J. ASTM Int., **8** 9 (2011).

[9] CONBOY, T. et al, Performance of an operating supercritical CO2 Brayton Cycle, J. Eng. Gas Turbines Power, **134** (2012) 12–1230.

[10] CONBOY, T., PASCH, J., FLEMING, D., “Control of a Supercritical CO_2 Recompression Brayton Cycle Demonstration Loop”, Proc. ASME Turbo Expo San Antonio, 2013, San Antonio, Texas, 2013.

[11] MOISSEYTSEV, A., SIENICKI, J.J., “Supercritical carbon dioxide Brayton Cycle for SFR applications: Optimization, transient analysis, and control”, paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[12] SIENICKI, J.J., “Modeling sodium oxide deposit growth and sodium plugging”, paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[13] LEE, C.H., YANG, W.S., “Verification and validation of Multigroup Cross Section Generation Code MC^2-3 for Fast Reactor system”, 2012 ANS Winter Meeting, (Proc. Conf. San Diego, 2012), American Nuclear Society, San Diego, CA (2012).

[14] FERENZ, R.M., HODGE, N.E., Adding a MOAB geometry interface to SHARP Structural Mechanics, Rep. LLNL-TR-558859, Lawrence Livermore Natl Lab., CA (2012).

[15] POINTER, D.W., et al. these Proceedings.

[16] SALVATORES, M. et al,, “WPEC SG33 on methods and issues for the combined use of integral experiments and covariance data”, Nuclear Data for Science and Technology (Proc. Conf. New York, 2013), BNL, New York (2013).

[17] BASGALL, M., Fission in a small package, Stewardship Science Magazine, 12, Krell (2010-2011).

[18] SOFU, T., “Parametric analysis of thermal stratification during the Monju Turbine Trip Test”, Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP) Chicago, IL, 2012, American Nuclear Societey, LaGrange Park, IL (2012).

[19] THOMAS, J.W., FANNING, T.H., DUNN, F.E., SOFU, T., “Analysis of the Phenix End-of-Life Natural Convection Test with SAS4A/SASSYS-1”, Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP), Chicago, IL, 2012, American Nuclear Society, LaGrange Park, IL (2012).

[20] SOFU, T., BRIGGS, L.L., “Benchmark specifications for EBR-II Shutdown heat removal tests” Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP), Chicago, IL, 2012, American Nuclear Society, LaGrange Park, IL (2012).

[21] DENMAN, M., LACHANCE, J., SOFU, T., FLANAGAN, G., WIGELAND, R., BARI, R., "Sodium Fast Reactor safety and licensing research plan", 2012 ANS Winter Meeting, (Proc. Conf. San Diego, 2012), American Nuclear Society, San Diego, CA (2012).

[22] PASAMEHMETOGLU, K., and J. CARMACK, these Proceedings.

[23] CHICHESTER, H.J.M., HAYES, S.L., PORTER, D.L., "Irradaition and postirradiation examination of AFC-1 transmutation metallic fuels for Fast Reactors", paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles (FR13), Paris, 2013.

[24] MCCLELLAN, K.J., HAYES, S.L., CHICHESTER, H.J.M., and VOIT, S.L., "Summary of the Minor Actinide-bearing MOX AFC-2C/D Irradiations", paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles (FR13), Paris, 2013.

[25] MARIANI, R.D., PORTER, D.L., BLACKWOOD, V., JONES, Z., OLSON, D., MISHRA, B., KENNEDY, J.R., HAYES, S.L., "New fuel alloys seeking optimal solidus and phase behaviors for high burnup and TRU burning", paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles (FR13), Paris, 2013.

[26] FIELDING, R.S., CRAPPS, J., UNAL, C., and KENNEDY, J.R., "Metallic fuel casting development and parameter optimization simulations", paper presented at Int. Conference on Fast Reactors and Related Fuel Cycles (FR13), Paris, 2013.

[27] FIELDING, R.S., KIM, K.H., GROVER, B., SMITH, J., KING, J., WENDT, K., CHAPMAN, D., ZIRKER, L., "Integral recycling test fuel fabrication", paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles (FR13), Paris, 2013.

[28] UNAL, C., CARLSON, N., GALLOWAY, J., "Preliminary simulation results of the constituent distribution model implemented into the BISON framework for the analysis of performance of metallic fuels", paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles (FR13), Paris, 2013.

[29] YUN, D., YACOUT, A.M., and CHEON, J. "Advanced multiphysics modeling of Fast Reactor fuel behavior", paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles (FR13), Paris, 2013.

[30] YACOUT, A.M., BILLONE, M.C., "Current status of the LIFE Fast Reactor fuel performance codes", paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles (FR13), Paris, 2013.

[31] SHEEN, S.H., GRANDY, C., et al, "Linear array waveguide transducer for undersodium viewing", paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles (FR13), Paris, 2013.

EVALUATION OF SEVERE EXTERNAL EVENTS ON THE JSFR

H. HAYAFUNE, A. KATO, Y. CHIKAZAWA, T. OHKUBO
Japan Atomic Energy Agency,
Oarai, Ibaraki

H. SAGAWA, Y. SHIMAKAWA
Mitsubishi FBR Systems, Inc.,
Shibuyaku, Tokyo

Japan

Abstract

The evaluation of earthquake and tsunami effects on the Japan Sodium-cooled Fast Reactor (JSFR) has been analysed. For seismic design, safety components are confirmed as maintaining their functions even against recent strong earthquakes. As for tsunami, some parts of the reactor building might be submerged, including the component cooling water system (CCWS) whose final heat sink is seawater. However, in the JSFR design, safety grade components are independent of the CCWS. The JSFR emergency power supply adopts a gas turbine system with air cooling, since the JSFR does not basically require quick startup of the emergency power supply owing to the natural convection decay heat removal system (DHRS). Even in the case of extended station blackout, the DHRS could be activated by emergency batteries or manually and be operated continuously by natural convection.

1. INTRODUCTION

The Japan Sodium-cooled Fast Reactor (JSFR) is a concept which has the potential to achieve the Generation IV reactor targets of sustainable energy production, radioactive waste reduction, safety equal to future light water reactors and economic competitiveness compared with other future energy sources. After the TEPCO Fukushima Dai-ichi nuclear power plant (1F) accident, external events became the focus of nuclear power plant design. In terms of robustness against external events, the JSFR had already improved safety features as a next generation reactor at a pre-conceptual design version in 2010, for example, passive shutdown system, core disruptive accident mitigation features and a passive decay heat removal system (DHRS). In this study, JSFR robustness against earthquake and tsunami has been evaluated, based on the 2010 design version. Additionally, an influence of an air cooler (AC) stack collapse due to severe external hazards had been studied.

2. JSFR VERSION 2010

2.1. General description

The JSFR plant concept of the 2010 version was basically established in the previous feasibility study from 1999 to 2006. Major specifications of the demonstration JSFR and system outline are shown in Table 1 and Fig. 1. The JSFR achieves the design requirements by adopting the following:

- High burnup fuel with oxide dispersion strengthened cladding material;
- Safety enhancement with self-actuated shutdown system and re-criticality free core;
- Compact reactor vessel (RV) adopting a hot vessel and in-vessel fuel handling with a combination of an upper inner structure (UIS) with a slit and advanced fuel handling machine;
- Two-loop cooling system with large diameter piping made of Mod. 9Cr-1Mo steel;
- Integrated intermediate heat exchanger (IHX)/pump component;
- Reliable steam generator with double-walled straight tubes;
- Natural circulation DHRS;
- Simplified fuel handling system (FHM);
- Containment vessel made of steel plate reinforced concrete;
- Advanced seismic isolation system.

TABLE 1. SPECIFICATIONS OF THE DEMONSTRATION JSFR

Items	Specifications
Electricity output	750 MW(e)
Thermal output	1765 MW(t)
Number of loops	2
Primary sodium temperature and flowrate	550/395ºC 1.62×10^7 kg/h/loop

TABLE 1. SPECIFICATIONS OF THE DEMONSTRATION JSFR (cont.)

Items	Specifications
Secondary sodium temperature and flowrate	520/335ºC 1.35 × 10^7 kg/h/loop
Main steam temperature and pressure	497ºC, 19.2 MPa
Feedwater temperature and flow rate	240ºC, 1.44 × 10^6 kg/h
Plant efficiency	Approx. 42%
Fuel type	TRU-MOX
Burnup (av.) for core fuel	Approx. 150 GW·d/t
Breeding ratio	Break even (1.03), 1.1, 1.2
Cycle length	26 m or less, 4 batches

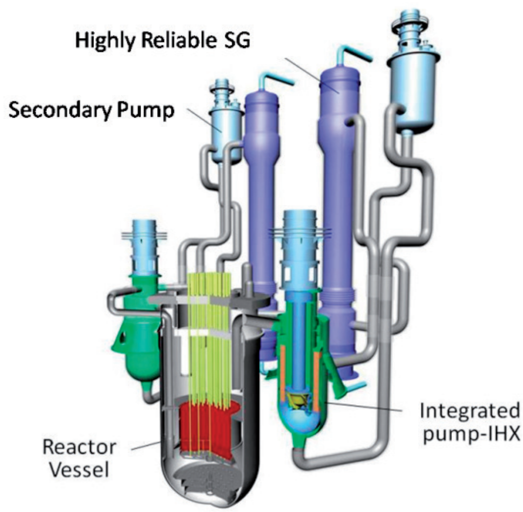


FIG. 1. JSFR system configuration.

As results of the FaCT Phase I, which was finalized in JFY 2010, the key technologies for the JSFR have been evaluated regarding whether those are feasible to be installed into the demonstration JSFR [1]. For seismic conditions, the JSFR updated design seismic load after the Niigata-ken Chuetsu-oki earthquake in 2007 [2]. The JSFR design seismic load was determined to cover the seismic loads in a typical site for an LWR at that time, with a certain margin in the natural period area of the principal components. The JSFR adopts an advanced seismic isolation system which mitigates the horizontal seismic force by thicker laminated rubber bearings with a longer period and the improvement of damping performance by adopting oil dampers. To confirm feasibility of the thicker laminated rubber bearings, a basic characteristic test with a 1:8 reduced scale model was carried out. As a result, the possibility of the application of the thicker laminated rubber bearings was confirmed [2].

2.2. DHRS

The JSFR DHRS consists of a combination of one loop of direct reactor auxiliary cooling system (DRACS) and two loops of primary reactor auxiliary cooling system (PRACS) adopting a full natural convection system (Fig. 2). The heat exchanger of DRACS is dipped in the upper plenum within the RV. The heat exchanger of each PRACS is located on the primary side upper plenum of an IHX. The DHRS can be operated by a fully passive feature with natural convection, which requires no active components such as pumps [3]. Thanks to the passive features, the frequency of protected loss of heat sink was evaluated to be 2×10^{-8}/reactor-year in the previous PSA analysis [4]. In the recent study, frequency of protected loss of heat sink has been re-evaluated, taking into account the recent JSFR design. The result shows that the frequency of protected loss of heat sink is lower than 10^{-8}/reactor-year taking into account accident management meeting requirements on frequency of core damage of 10^{-6}/reactor year and failure of containment function 10^{-7}/reactor-year [5].

2.3. Emergency power supply system

The JSFR emergency power supply system employs gas turbine generators with self-air-cooling instead of diesel generators, which are necessary to allow use of the auxiliary system, lubricating oil system and seawater cooling system. The most important reason for employing the gas turbine is the natural convection DHRS. With the natural convection DHRS, the JSFR does not rely on pumps in

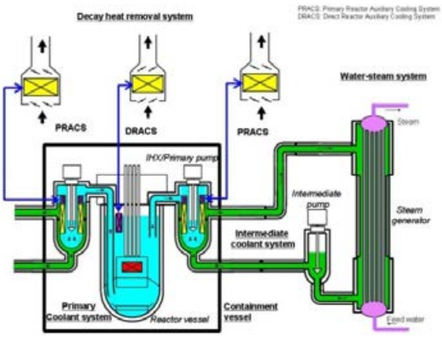


FIG. 2. DHRS configuration [6].

the main circuits during the decay heat removal operation. With a conventional forced circulation DHRS, there are pony motors on the main pumps. In the case of loss of an off-site power supply, the pony motors have to be activated within several seconds using diesel generators before completing pump coast-down. Since the JSFR does not require this kind of quick diesel generator initiation, it can employ gas turbine generators which require more than 30 seconds to be activated. In additional to the forced circulation DHRS, there are other safety components which require a power supply, including pumps for secondary decay heat removal circuits, blowers at the ACs and the component cooling water system (CCWS) for those safety components.

2.4. CCWS

The JSFR design features also enable mitigation of safety components in balance of plant (BOP). Table 2 shows a comparison between Monju and the JSFR in requirements on CCWS. In Monju, which adopted forced circulation DHRS, pony motor, electro-magnetic pumps (EMP) for the secondary DHRS, fuel handling components (cooling system of ex-vessel transportation machine (EVTM)) and the spent fuel pool require a CCWS with a safety grade. On the other hand, in the JSFR, no component requires CCWS as safety grade. In fact, there is no pony motor in the JSFR main circuits and EMPs in the secondary DHRS are non-safety grade owing to the natural convection DHRS. The JSFR utilizes an air conditioning system and an air cooling tower for cooling the EVTM and spent fuel pool, respectively, instead of CCWS. For emergency power supply, the JSFR adopts gas turbine generators which are able to self-cool using air.

TABLE 2. REQUIREMENTS ON CCWS

Items in CCWS	Monju	JSFR
Pony motor	safety grade	N/A
EMPs in DHRS	safety grade	non safety
Fuel handling	safety grade	air conditioning
Spent fuel pool	safety grade	air cooling
DG	safety grade	N/A

2.5. Fuel handling system

Figure 3 shows the configuration of the fuel handling system in the JSFR demonstration reactor. The JSFR adopts the TRU cycle with neptunium, americium and curium which cover the recycle of TRU elements recovered from LWR cycle. Since TRU compositions of LWRs would interfere with the decay heat of the fresh and the spent fuel subassemblies, design conditions of the fuel handling system were determined to cover the maximum decay heat. To achieve a good balance between handling such fuels and promising economic competitiveness, innovative designs to simplify the fuel handling system were introduced in the JSFR, for example, increased efficiency to shorten the refuelling period by transferring two spent fuel subassemblies at a time in the vessel, using a dry gas cleaning system to clean off the sodium to avoid sodium–water reaction in the spent fuel pool, etc. In terms of robustness against external hazards, the fuel handling system will keep their function thanks to the emergency on-site power supply, even when an earthquake occurs. Furthermore, the transport systems are connected to the uninterrupted DC power battery, which has the design capacity to run the transport systems and the valves for one hour. Hence, a fuel subassembly under handling can be transported to the RV, the EVST, or the spent fuel pool in a few minutes immediately after an earthquake and before a tsunami arrives at the JSFR.

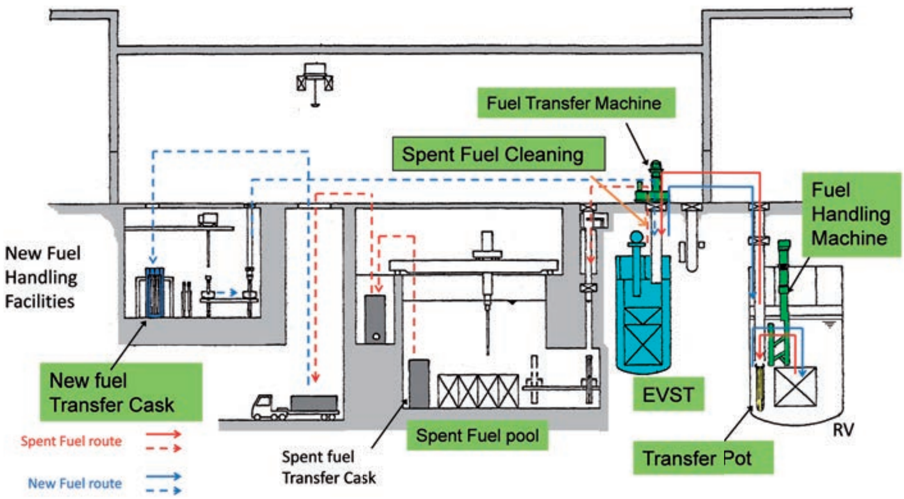


FIG. 3. Configuration of the JSFR fuel handling system.

3. SEISMIC EVALUATION

3.1. Analysis conditions

Seismic input at the reactor building basemat is shown in Fig. 4. In this study, a seismic condition which covers the acceleration and the velocity spectrum detected at the 1F NPP site (1F-envelope) was created based on the JSFR safe shutdown. The comparison has shown that the safe shutdown condition is more severe than the 1F-envelope condition. This means the JSFR safe shutdown condition in 2010 already accommodated the 1F earthquake conditions as it was. Using the 1F-envelope condition, the floor response spectrum has been analysed with the multi-mass model of the reactor building, as shown in Fig. 5. Since the JSFR employs a seismic isolation system, the calculational model takes them into account at the bottom of the reactor building. As an example, the floor response spectrum at the RV support is shown in Fig. 6. In the horizontal direction, acceleration of the JSFR is almost equal to the 1F-envelope condition. In the vertical direction, acceleration of the 1F-envelope condition is much less than that of the safe shutdown condition owing to the 1F ground condition which is input into the ground spring in the vertical model.

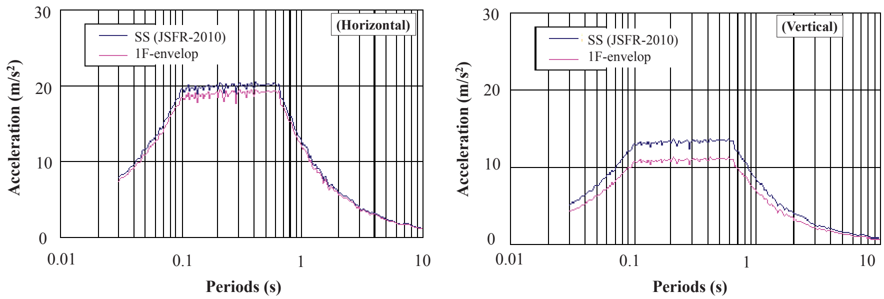


FIG. 4. Seismic input at reactor building basemat.

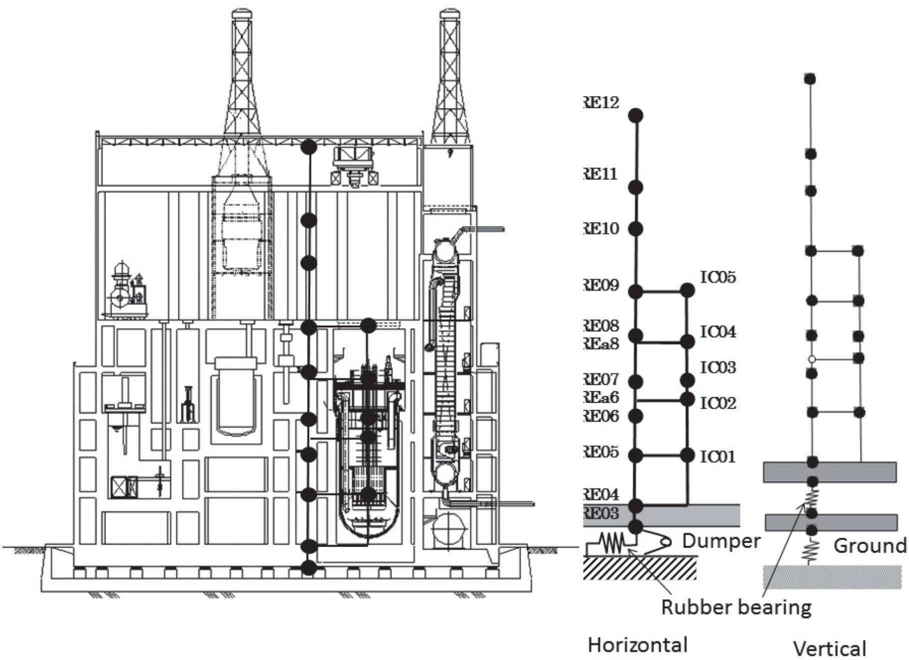


FIG. 5. Model for floor response analysis.

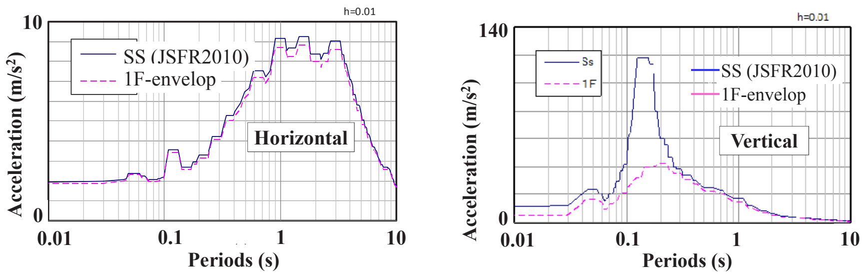


FIG. 6. Floor response spectrum at RV support.

3.2. RV

RV responses were calculated using the FINAS code [7] with the FEM using axis-symmetric shell elements. Calculation conditions and models are shown in Table 3 and Fig. 7. Analysed dominant vibration modes in horizontal and vertical directions are also shown in Fig. 7. Natural frequencies for the dominant modes are 11.0 Hz and 3.7 Hz for horizontal and vertical directions, respectively. Table 3 also shows analysed RV stresses. Buckling is evaluated with equations as follows [8]:

$$\left(\frac{Q}{Q_{cr}}\right)^5 + \left(\frac{M}{M_{cr}} + \frac{F}{F_{cr}}\right)^5 < \left(\frac{1}{f_B}\right)^5 \tag{1}$$

$$E_{Buck} = 1 \Bigg/ \left\{\left(\frac{Q}{Q_{cr}}\right)^5 + \left(\frac{M}{M_{cr}} + \frac{F}{F_{cr}}\right)^5\right\}^{1/5} \left\{\left(\frac{Q}{Q_{cr}}\right)^5 + \left(\frac{M}{M_{cr}} + \frac{F}{F_{cr}}\right)^5\right\}^{1/5} \tag{2}$$

where

Q	is the horizontal shear force;
Q_{cr}	is the critical shear buckling load,
M	is the bending moment;
M_{cr}	is the critical bending buckling moment,
F	is the axial compressive force;
F_{cr}	is the critical compressive buckling force;
f_B	is the safety factor (1.5 for the seismic condition);
E_{Buck}	is the buckling margin.

TABLE 3. RV CONDITIONS AND SEISMIC STRESS

Item	Value
Inner diameter	9460 mm
Height	20 700 mm
Thickness	50 mm
Material	316FR
Mass	2030ton
Shear stress	6.1 MPa
Bending stress	17.1 MPa
Axial compressive stress	27.8 MPa
Gravity stress	13.6 MPa

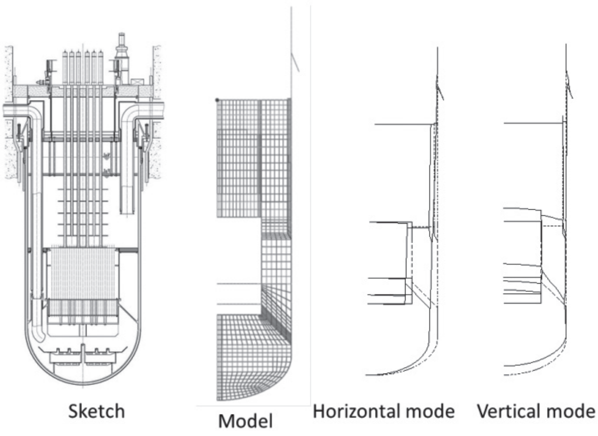


FIG. 7. Seismic analysis model of RV.

The evaluated buckling margin (E_{Buck}) is 3.7, while the required value is 1.5. The ratio between evaluated value and requirement is 2.4.

Parameters for control rod insertion evaluation are shown in Table 4. The natural frequency of the horizontal UIS vibration is evaluated as 3.6 Hz, showing that the displacement at the UIS bottom is 9.5 mm. Since the displacement at the core barrel top is 5.5 mm, relative displacement between the UIS bottom and core barrel top is 10.9 mm with root-mean-square. Limit relative seismic displacement is evaluated based on CR stuck displacement subtracted by displacement due to manufacturing tolerance, subassembly (SA) deviation and thermal expansion as follows:

$$\delta_{c'} = \delta_c - (\delta_1 + \delta_2 + \delta_3) \tag{3}$$

where

$\delta_{c'}$ is the limit of relative seismic displacement;
δ_c is the displacement to cause CR stuck;
δ_1 is the displacement due to tolerance;
δ_2 is the displacement due to SA deviation;
δ_3 is the displacement due to thermal expansion.

The ratio between the analysed and the limit (δ_C') values of relative seismic displacement is evaluated to be 1.7, thereby showing a safety margin.

TABLE 4. CR INSERTION CAPABILITY

Item	Value
Displacement at UIS bottom	9.5 mm
Displacement at core barrel top	5.5 mm
Relative seismic displacement with root-mean-square	10.9 mm
Limit of relative seismic displacement ($\delta_{C'}$)	19.1 mm
Seismic margin (=19.1 mm/10.9 mm)	1.7

3.3. Main components

Seismic evaluation of main components such as primary piping, etc., integrated IHX/pump, double-wall tube steam generator, EVST and EVTM:

- For primary piping, a beam model is employed (Fig. 8) taking into account guard and support pipes. The analysis has evaluated stresses at weak points such as nozzle at IHX, Y pieces, elbows and bell mouth, and results have been compared with the design limit stress of 250 MPa. Other piping has been evaluated in the same way.
- The analysis model for integrated IHX/pump is composed of a combination of shell and beam models (Fig. 9). For boundary security, buckling at the Y piece near the upper tube sheet and connection between shell and skirt are evaluated as high stress parts, and buckling margins are evaluated based on the demonstration reactor design standard [9] .
- The analysis model for the steam generator is composed of simplified beam models (Fig. 10). For heat exchange tubes and tie-rod, buckling and buckling margins are evaluated based on the demonstration reactor design standard [9].
- The analysis model for the EVST is the two-dimensional multi-mass model (Fig. 11). Connecting points around the upper plug are reconstructed by spring elements. The combined stress of the primary and the secondary stresses was estimated and compared with the corresponding criteria in the JSME, etc. The margin of seismic resistance is also estimated for the EVTM, the same way as for the EVST.

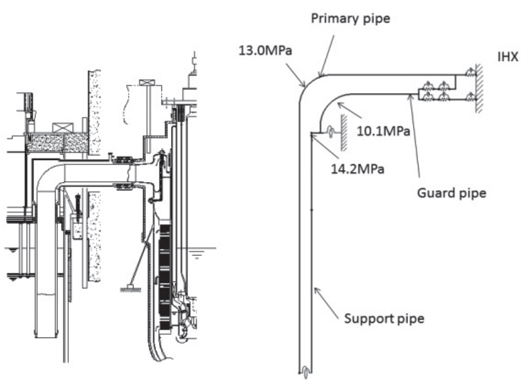


FIG. 8. Model for primary hot-leg piping.

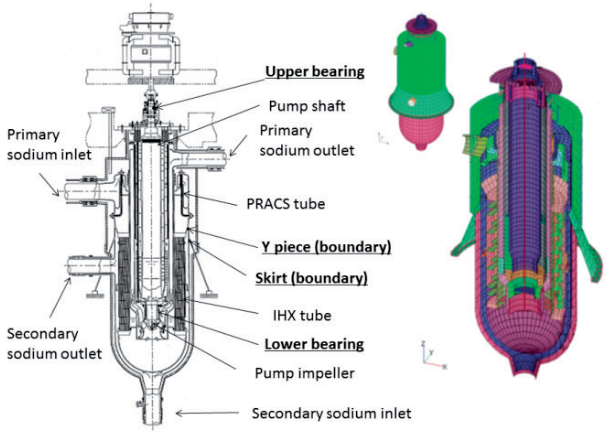


FIG. 9. Model for integrated IHX/pump.

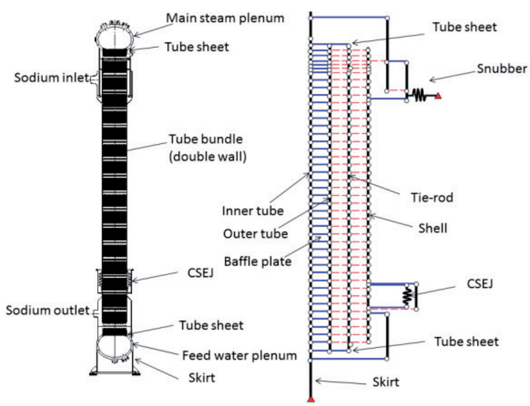


FIG. 10. Model for steam generator.

As the above seismic analyses show (Table 5), the integrity of the major components in the JSFR design version 2010 is confirmed against the 1F seismic condition. Furthermore, the JSFR is going to employ a new approach for external events [10]. From the viewpoint of a wide variety of external events consisting of natural and human-made hazards, representative events were selected in terms of siting, frequency, consequence, etc. For these representative events, design conditions were considered in the design base events and design extension conditions. The JSFR design will be improved to address these design basis events and design extension condition external hazards in a future study.

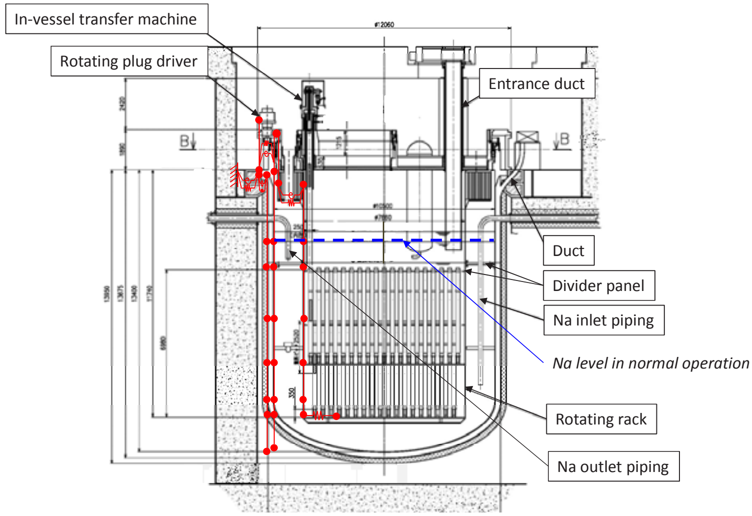


FIG. 11. 2-D multi-mass model for the EVST.

TABLE 5. SEISMIC EVALUATION RESULTS

Component	Item		Margin
RV	Buckling		2.4
	Control rod insertion capability		1.7
Integrated-IHX	Buckling	Y piece	22.8
		Skirt (support point)	50.0
Steam generator	Buckling	Skirt	21.1
		Main Body	20.9
Primary piping	Stress (hot-leg)	Support piping	17.6
	Stress (cold-leg)	Elbow center (outer piping)	5.3
Secondary piping	Stress (hot-leg)	Elbow canter	13.7
	Stress (cold-leg)	Elbow canter	47.2
DRACS secondary piping	Stress (hot-leg)	Elbow canter	4.7
	Stress (cold-leg)	Elbow canter	4.3

TABLE 5. SEISMIC EVALUATION RESULTS (cont.)

Component		Item	Margin
PRACS secondary piping	Stress (hot-leg)	T piping	3.9
	Stress (cold-leg)	T piping	4.7
EVST	Stress	Rotating rack	2.1
		Inner tank	2.5
		Outer tank	23.8
EVTM	Stress	Main body	23.4
		Rail hook	6.4
		Locator pin	1.9

3.4. Double pump seizure due to earthquake

The JSFR safety has been confirmed in case of pump seizure in one loop at rated operation as a design base accident [3]. Basically, pump seizure could not happen even in the JSFR shutdown system earthquake condition with a margin. For pump coast-down security, prevention of interaction between shaft and bearing has been evaluated in the 1F-envelope condition by response spectrum analysis. There are upper and lower bearings for the pump shaft. Evaluated seismic loads and criteria to prevent interaction between the shaft and bearing are shown in Table 6. Even though interaction between shaft and bearing can be prevented in the 1F-envelope seismic condition, as a hypothetical case, double pump seizure has been evaluated in terms of timing of double pump seizure from the earthquake detection. The analysis is carried out based on 1-D and detailed 3-D thermohydraulic analysis methods [11]: (1) conditions of the nominal hot fuel subassembly is evaluated by 1-D analysis, and (2) thermohydraulics in the hot fuel subassembly is analysed by the 3-D calculation. The criterion of coolant maximum temperature of 1020ºC is one of the design extension conditions criteria.

TABLE 6. EVALUATION OF INTERACTION OF PUMP SHAFT AND BEARING

Item	Upper	Lower
Evaluated load	74 kN	94 kN
Criteria to prevent interaction	260 kN	149 kN
Ratio of criteria to evaluated load	3.5	1.58

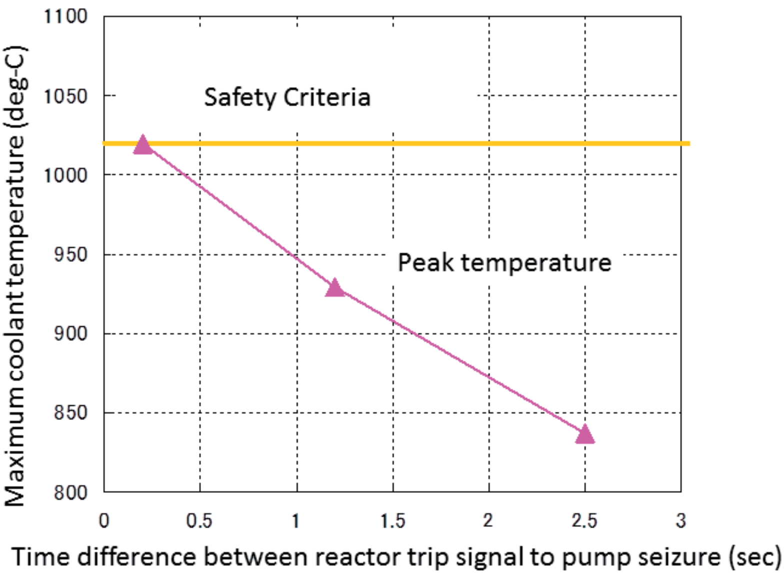


FIG. 12. Relation of pump seizure timing and maximum coolant temperature.

Figure 12 shows relation of double pump seizure timing and maximum coolant temperature. When a major earthquake occurs, the reactor trip signal is generated by detecting earthquake acceleration and the control rod is inserted in 0.2 s. In case 1 where the pump seizure happens at the same time of control rod insertion at 0.2 second, the coolant temperature is just under the criterion of 1020ºC. In fact, reactor shutdown would precede pump seizure because the seismic acceleration meter is directly located on the reactor building base while the major components are protected by seismic isolation. Regarding the time difference between shutdown and pump seizure owing to seismic isolation, the coolant boiling could be prevented.

In conclusion, as mentioned above, pump seizure can be prevented through the seismic design robustness, even if the 1F-envelope earthquake occurs. Additionally, the JSFR safety has been confirmed in the case of pump seizure in one loop as design base accidents. In the case of a hypothetical severe earthquake as a design extension condition, coolant boiling could be prevented when control insertion precedes double pump seizure. In the JSFR, there could be a time difference between earthquake detection at the reactor building basemat, which is a trigger for control rod insertion and response of components which are on the seismic isolation rubbers. These rubbers are installed on the reactor building basemat. Thus, JSFR safety could be confirmed in the case of hypothetical double pump seizure.

4. TSUNAMI EVALUATION

4.1. Evaluation assumptions

In this section, possible consequences caused by a tsunami are investigated. During the 1F accident, the seawater pumps for the CCWS could be damaged, since they are located at sea level. The CCWS could then fail, since it depends upon seawater as a final heat sink. As mentioned above, in the JSFR, safety components, including the DHRS and emergency power supply, are independent of the CCWS owing to full natural convection DHRS and air cooled gas turbine generators. The JSFR can then basically remove decay heat continuously by the natural convection DHRS.

On the other hand, since the JSFR is in a conceptual design phase, a site and its elevation for construction cannot be determined. Additionally, the tsunami height is dependent on the site conditions and seabed topography. Hence, in this study, the following factors were used to determine the potential characteristics of decay heat removal by natural convection:

- Hypothetical loss of the gas turbine generators immediately after the tsunami is assumed.
- Water flooding inside the reactor building is not assured. In fact, the uninterruptible DC power battery arranged in the reactor building would be assumed to survive.

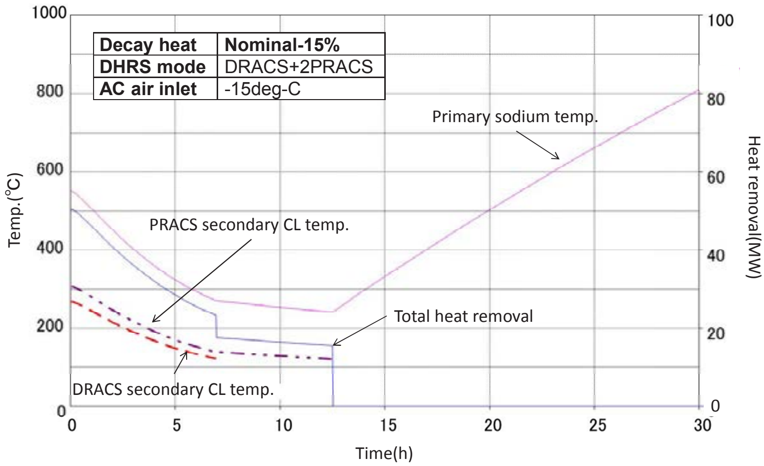


FIG. 13. SBO transient.

4.2. Hypothetical SBO analysis

In the case of a strong earthquake, the safety shutdown system and DHRS are activated immediately. In this study, hypothetical damage to the emergency power supply causing SBO is assumed. Since the JSFR's DHRS is a full natural convection type with adequate vertical intervals between the core to AC, as shown in Fig. 5, it can continue decay heat removal continuously even in a hypothetical SBO condition. However, the automatic control of DHRS could be lost after the power supply from the batteries is finished. In the 2010 JSFR design, the uninterruptible DC power battery can control automatically within 2 hours of event initiation. In this section, a transient in the case of SBO is analysed. In this analysis, criteria for boundary security and AC freezing are an RV temperature of 650ºC and an AC sodium outlet temperature of 120ºC. For AC freezing criteria, the limiting temperature means full freezing of AC. On the reactor trip, all DHRS, one DRACS and two PRACS are activated and DRACS/PRACS dumpers are controlled by power supply from the emergency batteries. After 2 hours, the DRACS and PRACS damper control is lost due to battery capacity and assumed to be maintained as is. In this case, the decay heat and AC air inlet temperatures are assumed to be low, investigating a typical AC freezing case. As a result, AC freezing in DRACS and PRACS happens in 7 hours and 12 hours after the reactor trip, respectively (Fig. 13). After the loss of DHRS, the RV temperature could reach 650ºC in 25 hours. This analysis suggests that JSFR has a certain time margin to cause protected loss of heat sink, even in the case of SBO; the time margin to DRACS AC freezing is 7 hours. Owing to the natural convection DHRS, required accident management is basically manual control of

AC dumpers by operators in the central control room. In fact, the AC design in the 2010 JSFR already accommodated manual operation of the AC damper and 7 hours is thought to be long enough for manual AC damper control.

Operation time is evaluated as shown in Table 7. The item ‘control room to AC’ includes operator movement, temperature measurement and decision time. Each AC has both inlet and outlet dampers and the dampers are doubled taking into account single failure. As shown in Table 7, operation times for DRACS and PRACS are evaluated to be 67 and 73 minutes, respectively. Those time margins show that the DHRS AC dampers are capable of being controlled manually with sufficient operation time compared with time margin of 7 hours as evaluated above.

4.3. SBO analysis in fuel handling system

Figure 14 shows the cooling system configuration of the EVST. Decay heat removal by natural convection is expected even during SBO since its final heat sink is air. The estimation is carried out applying the decay heat of fuel subassemblies immediately after the reactor shutdown. Figure 15 shows the temperature transient when the SBO occurs in normal fuel handling. EVST outlet temperature is saturated around 350ºC. Since the temperature criterion is set in terms of the integrity of the cladding material at under 550ºC for the long term, the EVST cooling system has sufficient cooling capacity by natural circulation.

TABLE 7. AC MANUAL OPERATION TIME

Item	Value
Control room to AC	37 min
DRACS AC inlet	5 × 2 min
DRACS AC outlet	10 × 2 min
PRACS AC inlet	6.5 × 2 min
PRACS AC outlet	11.5 × 2 min

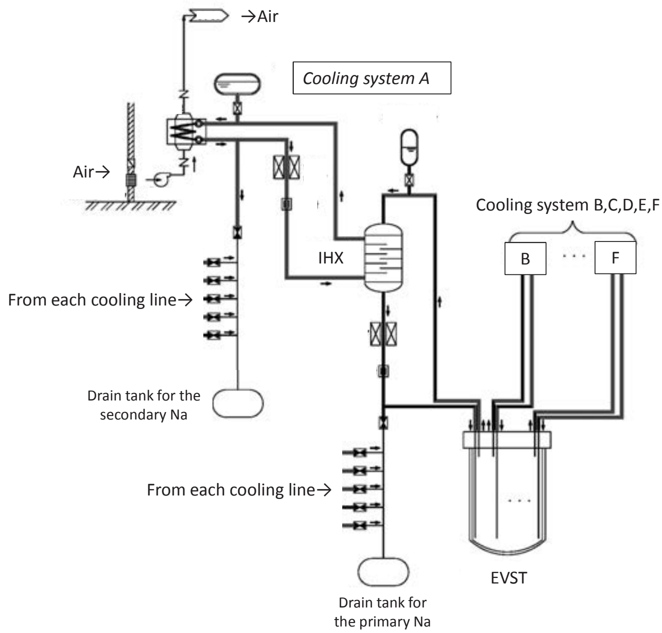


FIG. 14. Cooling system of the EVST.

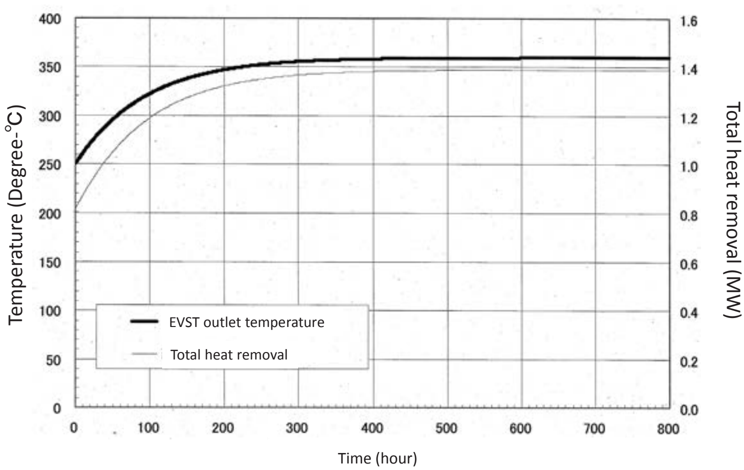


FIG. 15. Temperature transient of the EVST in the case of natural convection (normal fuel handling).

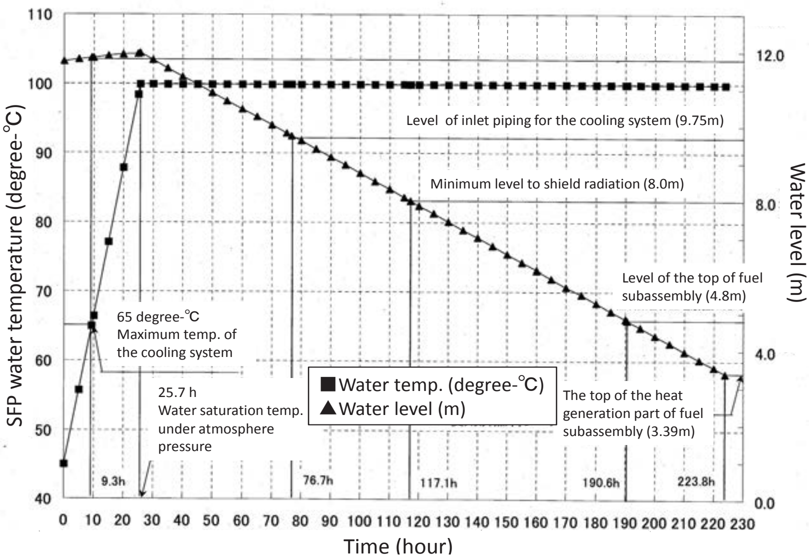


FIG. 16. Temperature transient of the spent fuel pool in the case of loss of emergency on-site power generator.

Water feeding is expected to stop when the SBO occurs since the pure water feeding system is not safety grade. Figure 16 shows the temperature transient and the water level of the spent fuel pool in the SBO. It shows that a sufficient time margin of 9 days is expected as the potential time elapsing before the water level reaches to the top of the heat generation part of the fuel subassemblies.

5. STACK COLLAPSE DUE TO EXTERNAL HAZARD

The protection of the stack for the DHRS air cooler is one of the most important things in the JSFR to ensure the final heat sink and to maintain draught power of natural convention. AC stacks are basically designed to stand against strong typhoons or winds. However, regarding design extension conditions, external hazards such as a severe typhoon, etc., a cooling capability of DRACS and PRACS ACs in the case of stack collapse has been evaluated. The cooling capability of DRACS and PRACS in natural convection mode with stack is 13.5 and 20.1 MW, respectively, and the requirement of decay heat removal is 18.4 MW. If one of PRACS AC stack is available, decay heat could be removed. In the case of stack collapse, the cooling capabilities decrease to 8.6 and 12 MW, respectively, since draught power of natural convection is reduced without stack. Even in an all stacks collapse, decay heat could be removed by maintaing air paths in two ACs by accident management. For the DRACS AC, a three-dimensional analysis using Star-CD has been conducted and the cooling capability evaluation has been validated. Figure 17 shows the calculational model and results. The results show that the DRACS AC has an 8.6 MW cooling capability, even in stack collapse, and the one and three dimensional calculation results are in good agreement. Responding to the result, measures are under discussion, as shown in Fig. 18. As shown, and since the main components of the AC is in the reactor building except for the stack in 2010 JSFR, the JSFR has the advantage of being able to allow accident management by operators and to add an air flow pathway.

6. CONCLUSIONS

JSFR resilience against earthquake and tsunami has been proved in the 2010 JSFR design version. Integrity of the major components has been confirmed covering recent earthquake conditions. In the case of the effects of tsunami on the JSFR, the consequences of a hypothetical SBO have been investigated. In the

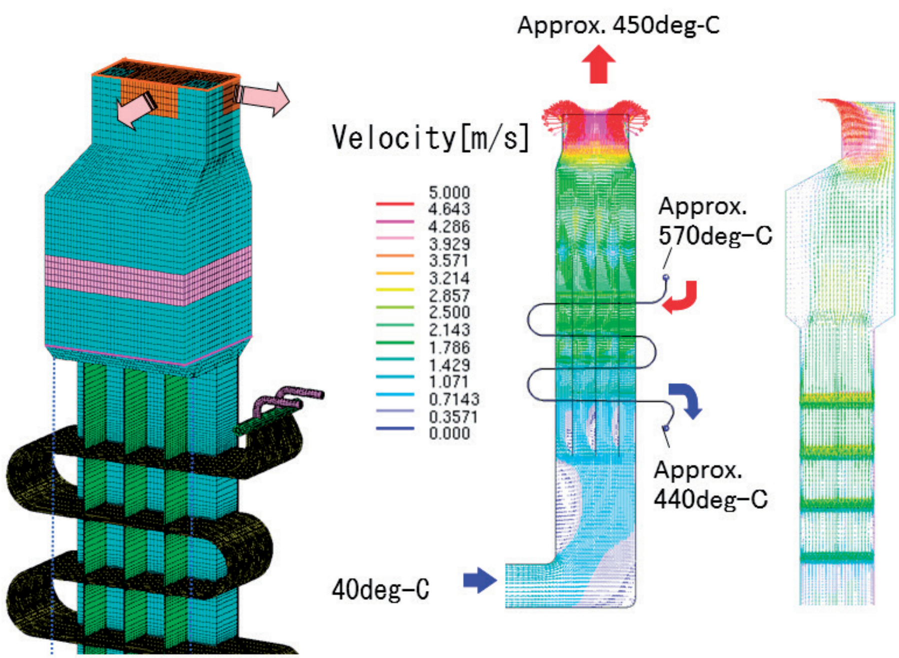


FIG. 17. AC analysis in the case of stuck collapse.

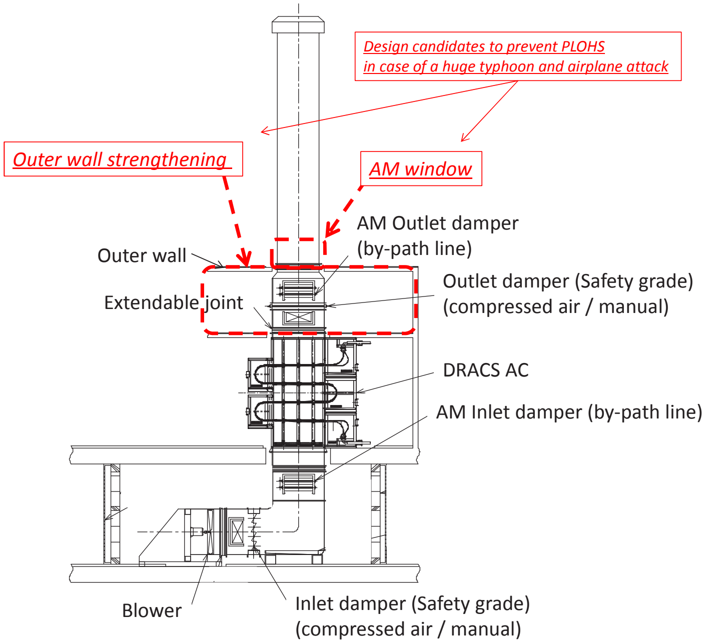


FIG. 18. Design measure candidates in AC.

case of SBO, decay heat is removed by natural convection DHRS. The analysis has revealed that AC freezing could happen due to loss of automatic control of AC dampers. However, the time margin to the AC freezing is evaluated to be 7 hours and to be enough for AM. Decay heat cooling in the case of collapse in all air stacks of ACs has been evaluated. Results show that decay heat could be removed by means of maintained air paths in two of three ACs by accident management. In conclusion, the JSFR in the 2010 design version has enough external hazard resilience, mainly due to passive safety features and the seismic isolation system. In the future, measures against all kinds of external hazards will be studied and enhanced in the JSFR design.

ACKNOWLEDGEMENTS

This paper includes results of the Technical Development Program on a Commercialized FBR Plant entrusted to the JAEA by the Ministry of Economy, Trade and Industry of Japan (METI).

REFERENCES

[1] AOTO, K., CHIKAZAWA, Y., KOTAKE, S., ITO, T., "Status of JSFR development in phase I FaCT Project", paper presented at Int. Conf. GLOBAL: Toward and Over the Fukushima Daiichi Accident, Makuhari, Japan, 2011.

[2] OKAMURA, S., et al., "Seismic isolation design for JSFR", 08-28P, paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), Kyoto, Japan, 2009.

[3] YAMANO, H., KUBO, S., et al., Technological feasibility of two-loop cooling system in JSFR, J. Nucl. Technol., **170** (2010) 159–169.

[4] KURISAKA, K., "Probabilistic safety assessment of Japanese sodium cooled Fast Reactor in conceptual design stage", paper presented at 15th Pacific Basin Nuclear Conference (PBNC-15), Sydney, Australia, 2006.

[5] CHIKAZAWA, Y., et al., paper 11278 presented at Int. Congress on Advances in Nuclear Power Plants (ICAPP), Nice, France, 2011.

[6] KUBO, S., et al., "Safety design requirements for safety systems and components of JSFR", 03-10, paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), Kyoto, Japan, 2009.

[7] MORISITA, M., ''Seismic response analysis of FBR core by FINAS code'', paper presented at 14th Int. Conf. on Structural Mechanics in Reactor Technology (SMiRT-14), Lyon, France, 1997.

[8] KAWASAKI, N., et.al., "Recent design improvements of elevated temperature structural design guide for DFBR in Japan", Div.F F04/4, paper presented at 15th Int. Conf. on Structural Mechanics in Reactor Technology (SMiRT15) , Seoul, Korea, 1999.

[9] MATSUURA, S., et al., JSME Int. J. Ser. B, Fluids and Thermal Eng. **36-B** (3) (1993) 485–492.

[10] YAMANO, H., "Safety design approach for external events in Japan sodium-cooled Fast Reactor", Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP), Chicago, IL, 2012, American Nuclear Society, LaGrange Park, IL (2012).

[11] OHSHIMA, H., "Development of core hot spot evaluation method for decay heat removal by natural circulation under transient conditions in sodium cooled fast reactor", Transactions of the Japan. Soc. of Mechanical Engineers, Series B, **76** 763 (2009) 86–88 (in Japanese).

FAST REACTOR SAFETY: POST-FUKUSHIMA LESSONS AND GOALS FOR NEXT GENERATION REACTORS

Chairpersons

P. MARITEAU
France

A. YAMAGUCHI
Japan

MODERN APPROACHES TO SAFETY ASSURANCE OF A NEW GENERATION OF SODIUM FAST REACTORS

I.M. ASHURKO, A.A. KAMAYEV, V.M. POPLAVSKIY
FSUE "SSC RF — Institute for Physics and Power Engineering",
Obninsk

B.A. VASILYEV, S.L. OSIPOV, A.I. STAROVEROV, S.F. SHEPELEV
JSC "OKBM Afrikantov",
Nizhny Novgorod

Russian Federation

Abstract

In the stage of designing sodium cooled fast reactors (SFRs) of a new generation there is a task to improve their inherent safety up to the level higher than that of the previous SFR designs. Modern safety requirements to the SFR of the fourth generation are described. Through the example of the BN-1200 reactor, approaches to safety assurance are demonstrated using development of inherent safety properties up to a brand new level compared to that of the earlier commercial reactor designs (BN-600 and BN-800). Also, passive safety devices and systems applied in the BN-1200 design are described. The goal is to eliminate the necessity for evacuation of residents under conditions of any possible realistic accidents. The paper presents properties of inherent self-protection of the BN-1200 reactor and estimation of their effectiveness in terms of safety assurance. The basic design approaches concerning safety are considered, including additional measures as applied to the BN-1200 reactor. These include measures aimed at the elimination or minimization of sodium leaks; design approaches to the passive shutdown systems (PSS) using various operation principles, namely: hydraulically suspended absorber rods operating in case of coolant flow rate decrease (PSS-H) and absorber rods operating in the case of increase of the core outlet coolant temperature above a certain value (PSS-T); passive decay heat removal system; sodium plenum above the core; gastight compartment above the reactor; core catcher made of refractory metal; and reactor guard vessel.

1. INTRODUCTION

With the depletion of mineral resources (oil, gas and coal) against the background of increasing societal demand for electrical energy, the global environmental and climatic problems related to the use of fossil fuel power plants, as well as the costliness and insignificant numerical strength of renewable energy sources has caused renewal of interest by both community and industry in nuclear power, its capabilities being sufficient for assuring the energy supply of society for many hundreds years to come. In this view, the Fukushima-1 accident has shown that nuclear power can only have a chance for its development and be accepted by the community if the safety problems are comprehensively solved in order to eliminate serious negative environmental, social and economic consequences in cases of severe accidents in power plants.

Safety assurance for the advanced nuclear energy systems (NES) is a top priority and urgent problem.

Sustainable development of nuclear power in the future can only be assured by means of fast reactors, which would facilitate expansion of the fuel basis of nuclear power by the use of uranium-238. In this view, it is supposed that fast reactors of the fourth generation would make the basis for future nuclear power. The following fourth generation reactor types are considered as candidates:

— Sodium cooled fast reactors (SFRs);
— Fast reactors with heavy liquid metal coolants (HLMC);
— Gas cooled fast reactors;
— Supercritical water cooled reactors;
— Molten salt reactors.

SFRs have become the most developed and mastered type compared to the other fast reactors in the list.

The new generation SFR designs are now under development in several countries (China, EC, France, India, Japan, Republic of Korea and the Russian Federation), where related national programmes of nuclear power development have been embraced.

This paper is devoted to considering modern approaches to the assurance of safety of advanced SFR designs.

2. SAFETY REQUIREMENTS OF FOURTH GENERATION NUCLEAR ENERGY SYSTEMS

All the above fast reactor types are taken into consideration and studies within the framework of the Generation-IV International Forum (GIF) are consolidating efforts of the main countries developing reactor technologies, namely: Argentina, Brazil, Canada, China, EC, France, Japan, Republic of Korea, South Africa, Switzerland, the Russian Federation, the United Kingdom and the United States of America [1].

Within the framework of the GIF the goals which should be achieved in the fourth generation NES designs have been formulated in the most general way. These include the following safety requirements to be met by the fourth generation reactor technologies [2]:

— Safety and reliability should exceed the level in existing NES.
— The probability and degree of reactor core damage should be minimized to a very low value.
— The need for any off-site emergency response should be eliminated.

In the course of specification of safety requirements, the specialists have come to the conclusion that it is necessary to work out safety design criteria, taking into account specific features for each type of fourth generation NES.

For this purpose, the special task force was set up within the GIF framework, which developed safety design criteria for SFRs, which represents the most mastered technology among the various FR types.

3. SFR FEATURES

As it has been mentioned above, development of safety requirements and approaches to these requirements for each NES should be carried out taking into account specific NES features.

In this view, it is worthwhile to recall the basic specific features appropriate to SFRs.

In fact, there is only one noticeable drawback of the SFR, namely, chemical activity of sodium coolant with respect to air and water, which should be taken into account when considering leaks in sodium systems, in particular, tube leaks in the steam generators (SGs) in the case of application of water and steam in the tertiary circuit.

In addition, SFR safety analysis should include study on sodium coolant activation (formation of radioactive isotope sodium-24 with ~15 hours half-life) and the possibility of realization of a positive sodium void reactivity effect in the case of sodium boiling onset in the core.

Among SFR advantages there are inherent safety characteristics and properties presented in the following.

(1) Fair thermophysical characteristics of sodium coolant, which are qualified as inherent safety characteristics, namely:
   - High thermal conductivity of sodium would assure high heat transfer in both forced and natural coolant flow modes and, hence, the relatively small-size reactor core with the stable, easy-to-control power profile can be used;
   - Rather low melting point (98°C), making it possible to easily maintain liquid coolant condition;
   - High boiling temperature (880–960°C), depending on pressure in the circuit) assures high level of working temperatures (500–550°C) with high margin (over 300°C) for sodium boiling temperature;
   - High thermal capacity of the sodium circuits would facilitate smoothing of transients under conditions of an unprotected loss of heat sink accident;
   - The possibility of providing effective natural circulation of sodium coolant in the heat removal circuits assures self-protection against the accidents with loss of power supply;
   - Experimentally confirmed possibility of stable effective heat removal from the fuel pins in the coolant boiling mode.

(2) Good sodium compatibility with structural materials and its low corrosiveness.

(3) The possibility of system operation under low pressure in the circuits (close to atmospheric pressure value) within the working temperature range.

(4) Negative temperature and power reactivity effects are typical for a SFR core, assuring stable negative power and temperature reactivity feedback.

Historically, SFR designs in the different countries have had a similar three-circuit heat removal system including the primary and the secondary sodium circuits and the tertiary circuit with water coolant. This approach makes it possible to eliminate the interaction of radioactive sodium with water in the case of SG leaks, as well as eliminate overpressure of the primary circuit under conditions of such leaks.

The primary circuit configuration can be either pool or loop type, each option having its specific effect on safety. For instance, the addition of a guard reactor vessel over the main reactor vessel in pool reactor design makes it possible to practically eliminate radioactive sodium leaks from the reactor.

4. SFR DEVELOPMENT IN THE RUSSIAN FEDERATION

In the Russian Federation, significant experience gained in SFR development and operation has demonstrated a high degree of assimilation and reliability of this reactor technology, which has implications for achieving a high safety level for advanced SFR designs.

In total, four SFRs have been successfully operated in the former USSR and in the Russian Federation:

(1) Experimental reactor BR-5/10 in Obninsk (start-up on 26 January 1959, finally shut down on 6 December 2002);
(2) Research reactor BOR-60 in Dimitrovgrad (Melekess) (startup on 14 December 1969, its lifetime was extended until 31.12.2014);
(3) Prototype power reactor BN-350 in Aktau (Shevchenko) (first criticality gained on 29 November 1972, finally shut down on 22 April 1999);
(4) Commercial power Unit No. 3 of Beloyarsk NPP with BN-600 reactor (first criticality gained on 26 February 1980, a 30-year design lifetime has been extended up to 40 years, until 31 March 2020).

As of December 2012, the total SFR operational experience in the former USSR and the Russian Federation covers more than 147 reactor-years, i.e. over 34% of the worldwide SFR operational experience (over 427 reactor-years).

Generally, gained operating experience can be considered successful, demonstrating the possibility of the safe and cost effective operation of the SFR. For instance, the average load factor of No.3 power unit of the Beloyarsk NPP with the BN-600 reactor during its commercial operation since 1983 is over 74% [3].

As regards safety issues concerning sodium coolant specific features, it should be noted that the problem of coolant leaks from the circuits and in SGs has been successfully solved.

On the basis of the experience gained on sodium leaks in the circuits and SG tubes in the early stage of sodium technology development [4, 5], safety systems were designed that were capable of preventing incidents with failures of sodium circuits at the earliest stage of their development and, by this, eliminating any negative consequences. It should be recalled that the last sodium leak in the circuit of the BN-600 power unit occurred in May 1994, and the last SG leak occurred in January 1991 in spite of multiple replacements of SG modules carried out after that event in accordance with the regulations in the course of the BN-600 lifetime extension [6].

Nowadays, advanced fast reactor designs to be used as the new technological platform for future nuclear power in the Russian Federation are developed within the framework of the Federal Target-oriented Program (FTP): "Nuclear Power Technologies of a New Generation for the Period 2010–2015 and the Outlook to 2020" adopted in the early 2010s. According to the FTP, work is carried out in two areas, namely: SFRs and fast reactors with heavy liquid metal (lead or lead–bismuth) coolant.

As regards the SFR line, this is the development of a design of the commercial large size power unit (BN-1200 reactor), which is considered as the fourth generation NES. The basic design of the BN-1200 power unit is planned to be finished by 2015. Now, the discussion is under way on the possibility of construction of a head power unit with the BN-1200 reactor at the Beloyarsk NPP site.

Based on the positive experience gained with SFR operation in the Russian Federation, the decision to use a traditional three-circuit design with the steam–water tertiary circuit was made as applied to the BN-1200 reactor.

The following measures are planned for justification of the BN-1200 power unit design:

— Expansion and upgrading of the experimental facilities fleet in the nuclear power area (construction of the new MBIR research reactor, upgrading of the fast critical facilities (BFS), completion of construction of the SAZ test facility for investigation of SG protection systems, creation of other new test facilities and upgrading of existing facilities);
— Development of the new generation computer codes for justification of parameters and safety of the BN-1200 power unit.

In addition to justification of the safety systems of the BN-1200 reactor facility and reactor safety characteristics, the updated experimental facilities will be used for the experimental studies required for verification and validation of the new computer codes for safety analysis.

5. APPROACHES TO THE BN-1200 SAFETY ANALYSIS

The BN-1200 design developed within the framework of the FTP is an integral part of the ‘Breakthrough’ project aimed at the creation of the new technological platform based on fast neutron reactors and closed nuclear fuel cycle. This design implies implementation of the natural safety principle assuming application of inherent safety properties to the maximum extent and, hence, allowing deterministic elimination of severe accidents that require evacuation of the inhabitants from the area adjacent to the NPP [7].

It should be noted that BN-1200 power unit design development, including choice of the main characteristics (Table 1) and design approaches, in particular, those on safety systems, is carried out taking into account all experience gained in SFR design, construction and operation (BR-5/10, BOR-60, BN-350, and primarily BN-600 and BN-800).

TABLE 1. MAIN BN-1200 PARAMETERS

Parameter	Value
Rated thermal power (MW)	2800
Electric power (MW)	1220
Load factor (%)	90
NPP efficiency (%):	
— gross	43.5
— net	40.7
Number of heat removal loops	4
Primary and secondary circuit coolant	Sodium
Working medium in the tertiary circuit	Water/steam
Design lifetime (years)	60
Flowrate of the primary sodium (kg/s)	15 784
Flowrate of the secondary sodium (kg/s)	12 776
Primary sodium temperature (IHX outlet/inlet) (°C)	410/550
Secondary sodium temperature (SG outlet/inlet) (°C)	355/527
Tertiary circuit parameters:	
— live steam pressure (MPa)	17.0
— live steam temperature (°C)	510
— feedwater temperature (°C)	275
— steam reheating option	Steam
Fuel	Mixed nitride and MOX fuel

The task has been stated to improve inherent safety properties (‘inherent self-protection’, according to the terminology adopted in the OPB-88/97 regulatory document [8]) up the higher level as compared to that of the BN-600 and BN-800 designs, and also to use additional passive systems and devices, all this being aimed at the assurance of meeting the following requirements under all possible realistic conditions:

- Making the reactor subcritical with the temperatures of its components maintained at the acceptable level;
- Decay heat removal from the reactor without any damage to its structures;
- Confinement of the major part of radioactive products released from the reactor under possible accident conditions within the reactor building.

Implementation of all these measures would assure achievement of the stated goal of avoiding inhabitant evacuation under any accident conditions in the BN-1200.

In the course of design studies concerning the choice of the BN-1200 reactor characteristics and options for safety systems, as well as reactor safety assurance issues, priority is placed on the use of SFR inherent safety properties and safety system designs based on passive operation principles.

The most important new design approaches used in the BN-1200 for its safety improvement as compared to those applied for the previous Russian SFR designs are as follows:

- Pool design of the primary circuit implying location of all sodium systems (including cold traps, neutronics and chemical engineering control systems) in the reactor vessel. This practically eliminates the possibility of primary radioactive sodium release from the reactor and its fire (Fig. 1).
- Transition from sectional–modular SG design to an integral one based on the application of the large size straight tube SG modules, thus significantly decreasing the surface area of sodium system components.
- Technical decisions on passive shutdown systems (PSS) based on using various operation principles, namely: hydraulically suspended absorber rods operating in the case of coolant flowrate decrease in the core (PSS-H) and absorber rods operating in the case of increase of the core outlet coolant temperature above a certain value (PSS-T).
- Passive decay heat removal system (DHRS) using independent loops connected directly to the reactor vessel.
- Design approaches to elimination or minimization of non-radioactive sodium leaks from the secondary loops and intermediate DHRS loops.
- Gastight compartment above the reactor used for radioactivity confinement in the case of a severe accident.

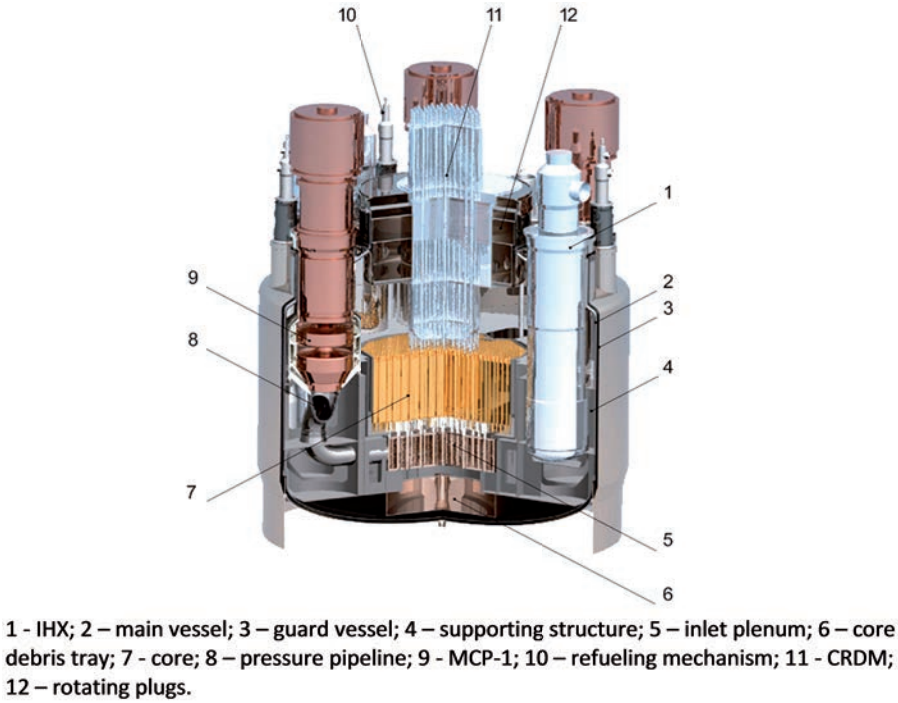


FIG. 1. Layout of the BN-1200 primary circuit.

In addition, the following design approaches, which were tested as applied to the BN-800 reactor design, are used in the BN-1200 design:

— Sodium plenum above the core for decreasing sodium void reactivity effect;
— Core catcher made of refractory metal for confinement and cooling of corium in the case of a postulated core disrupture accident, thus eliminating release of molten core debris from the reactor vessel;
— Reactor guard vessel preventing radioactivity release from the reactor even in the case of failure of the main reactor vessel.

This approach to safety assurance combined with the defence-in-depth principle implemented in the BN-1200 design would allow meeting both requirements of the Russian safety regulatory documents now in force and those imposed up to the fourth generation advanced reactors.

For instance, the BN-1200 design should meet the following requirements:

- The total probability of severe beyond design basis accidents that can lead to significant damage or meltdown of the reactor core should not exceed a value of 10^{-6} 1/reactor-year and all damaged structural elements of the core should be kept within the reactor vessel.
- The value of radioactivity release into the environment under any possible realistic accident should not exceed the radiation dose for the inhabitants outside the NPP site boundaries, specified by the regulatory documents, that requires evacuation of residents. The term ‘possible realistic accident’ means any initial event, even with very low likelihood, accompanied by superposition on the initial event of: failures of all active safety systems, single failures of elements of passive safety systems (having mechanical moving parts) provided by the design for restriction of consequences of the given initial event, failures of active safety related systems of normal operation, and erroneous actions of the personnel.

In the course of the BN-1200 safety analysis, it is planned to use both deterministic and probabilistic methods. For instance, severe accident analysis is made using a conservative approach, i.e. it is assumed that the initial event is accompanied by the failure of all active design safety systems and by the additional single failures in available passive safety systems.

Moreover, in order to evaluate the degree of self-protection of the BN-1200 against severe beyond design basis accidents, tentative studies were made on the ULOF accident with postulated failure of all active and passive reactor shutdown systems including all PSS-H and PSS-T rods. The preliminary results of these studies show the possibility of avoiding melting of fuel pin claddings and fuel even in the case of sodium boiling onset in the core. This is provided by the use of special design of the BN-1200 reactor core with a sodium plenum at the core outlet. Besides, alternative core design options, such as an axial layer of depleted uranium, layer with moderator, etc., are studied in order to decrease the sodium void reactivity effect value.

Chosen passive DHRS design increases fundamentally the resistance of the BN-1200 power unit against the accidents similar to those that occurred in Fukushima-1 NPP, since it does not require any additional power source for its operation.

The chosen arrangement of the primary circuit of the BN-1200 allowed elimination of all radiological consequences of sodium leaks in the system.

6. NECESSITY FOR INTERNATIONAL COOPERATION IN THE SFR SAFETY AREA

It is shown by the experience of severe accidents that have occurred in the nuclear facilities that all reactor technologies should meet the essential general set of safety requirements in spite of specific features of their development in the different countries and specific approaches to safety assurance.

In this view, there is a pressing need for the development of common safety criteria and requirements to be met by the fourth generation NES, including those with SFRs.

This activity has been already initiated as development of safety design criteria for SFRs within the GIF framework. For this purpose a special task force was set up with the participation of representatives from various countries (China, France, Japan, Republic of Korea, Russian Federation and the USA) and organizations (EC and IAEA).

The IAEA would play an important role in coordination of efforts by the international community in the development of common approaches and requirements to safety assurance of the fourth generation NES, including those with SFRs.

7. CONCLUSIONS

Analysis of approaches to safety assurance for advanced SFR designs of the next generation made with regard to the BN-1200 design, which is now under the development in the Russian Federation, shows the possibility of meeting all safety requirements imposed up to the fourth generation NES.

This makes it possible to consider SFRs as the real basis for achieving the goal of sustainable and safe development of nuclear power even in the nearest future.

The international community is conscious of the need for the development of the common safety standards as applied for the fourth generation NES including those with SFRs taking into account operational experience gained in all types of NES.

REFERENCES

[1] OECD NUCLEAR ENERGY AGENCY, GENERATION IV International Forum (GIF) 2011 Annual Report, OECD Publishing, Paris (2012).

[2] OECD NUCLEAR ENERGY AGENCY, GENERATION IV International Forum (GIF) 2007 Annual Report, OECD Publishing, Paris (2008).

[3] BAKANOV, M.V., NOSOV, Y.V., POTAPOV O.A., "Operational experience of the sodium-cooled Fast Reactor BN-600", paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles (FR13), Paris, 2013.

[4] POPLAVSKY, V.M., et al., "Review of Fast Reactor operational experience gained in Russia, approaches to coordinated research project", Operational and decommissioning experience with fast reactors, IAEA-TECDOC-1405, IAEA, Vienna (2004) 107–134.

[5] ASHURKO, Y.M., et al., "Fast Reactor operating experience gained in Russia: Analysis of anomalies and abnormal operation cases", Unusual occurrences during LMFR operation, IAEA-TECDOC-1180, IAEA, Vienna (2000) 117–144.

[6] SARAEV, O.M., et al., "Experience gained in Russia on sodium cooled Fast Reactors and prospects of their further development", in Proc. Int. Conf. on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), Kyoto, Japan, 2009, IAEA, Vienna (2012) 363–381.

[7] VASILYEV, B.A., et al., Implementation of the Inherent Safety Principle in the BN-1200 Design, J. Nucl. Environmental Saf., No. 1 (2012).

[8] General Regulations on Ensuring of Nuclear Power Plants Safety OPB-88/97, NP-001-97 (PNAE G-01-011-97), Moscow (1997).

SAFETY UPGRADING OF THE PFBR IN THE WAKE OF THE FUKUSHIMA ACCIDENT: SEVERE ACCIDENT MANAGEMENT STRATEGIES

P. KUMAR
Bharatiya Nabhikiya Vidyut Nigam Limited (BHAVINI),
Department of Atomic Energy,
Kalpakkam, India

Abstract

The Fukushima events have been reviewed and actions have been taken by many nuclear plants worldwide. The Prototype Fast Breeder Reactor (PFBR) is a coastal plant. It required thorough review, stress test and improvements in some areas. The paper will bring the details of the review and upgrading undertaken. The PFBR had faced a tsunami in December 2004, and as a consequence, changes were made in the design to accommodate the water level rise during a tsunami. Fukushima provided additional feedback on design features to face severe natural events. PFBR has made preparations to counter the effects of severe natural events. Additionally, preparedness is directed towards quick decision making, fast response time, training of staff to counter the effects of an accident, stores for emergency, approach route, psychological factors, etc., are addressed. Factors that were reviewed and underwent tightened control included ensuring reactor shutdown and maintenance of shutdown state; ensuring core and spent fuel cooling; avoidance of water entry into any vital areas that could affect functioning of important equipment; training of operators to respond to a situation not entirely conceived in advance; strengthening of support systems for operators to act swiftly; and management actions to address psychological issues associated with calamities

1. INTRODUCTION

The Fukushima incident shook the global nuclear community. Common cause failure in multisystems and in multiple units was never envisaged before Fukushima. The safety criteria required a thorough review. Not much change in safety philosophy was possible for the operating reactors worldwide. Back fits for core cooling and alternative power supply was the option adopted by utilities worldwide to strengthen safety.

2. INDIAN OCEAN TSUNAMI EFFECT ON PFBR

BHAVINI is a public sector company responsible for the construction and operation of fast reactors. Currently the first commercial fast reactor, the Prototype Fast Breeder Reactor (PFBR) is under construction. Many more are planned. The PFBR achieved 90% physical progress by the end of September 2012 and is moving towards sodium transfer in the secondary system loops. The project faced the tsunami on 26 December 2004 when the second pour of nuclear island raft concreting was in progress. The excavated pit for construction of nuclear island buildings was completely inundated. The inrush of seawater carried all plant, machinery, equipment, materials available at site into the pit and buried them under 3.5 m of sludge and 3.5 m of water overlying the sludge. Several modifications were done before reconstruction was commenced. This included:

— Creation of sea wall on the beach to avoid beach erosion during future tsunamis and cyclones.
— Creation of engineered outfall structure parallel to the coast at a distance of 1.8 km before it opens into the sea.
— Creation of a tsunami wall, made up of stone boulders to provide a barrier to break the energy of any future tsunami water. The height of the tsunami wall was raised to 5.1 m above mean sea level although the water rise during the 26 December 2004 tsunami was measured at 4.1–4.8 m at different places on the site.
— The nuclear island’s finished grade level was raised to 9.2 m and the finished floor at 9.5 m above mean sea level. This required modification to the layout. The nuclear island, power island, site assembly shop administration block and approach roads to the site ended up at different levels as a result of the nuclear island level raise. The nuclear island level in the process was 2.5 m above the power island. With the power island finished, the floor level was around 2.3 m above the highest water mark in the 2004 tsunami, and since the power island does not house safety systems, no further action on the power island against tsunami was considered warranted.

3. UPGRADING THE PFBR AFTER FUKUSHIMA

Factors that were reviewed after Fukushima included:

— Ensuring reactor shutdown and maintenance of shutdown state in the event of natural calamity. Robust design features of the PFBR, including the passive heat removal system, were found sufficient to shut down, maintain

shutdown state and remove decay heat from the reactor for an extended period, without any limit on time.
- Ensuring core and spent fuel cooling.
- Prevention of water entry into any vital areas that could affect functioning of important equipment.
- Training of operators to respond to emergency handling, emergency response, informing appropriate government bodies for external support and also to report any unforseen situation not conceived in advance.
- Strengthening of support systems for operators to act swiftly.
- Management actions, including addressing psychological issues associated with calamities.

From the Fukushima incident it was realized that the earthquake and tsunami exceeded design limits. The height of water rise due to the tsunami was different at different places along the same coast. Nuclear experts from Japan have spoken of the need for future design to take account of a major tsunami.

Observations have indicated that the height of tsunami water level at any point could be the result of a combination of factors. These include:

- Bathymetry of sea bed;
- Shore contour;
- Resistance to water on the shore;
- Shore levels/profile;
- Wave pattern when striking the shore;
- Estuaries, etc.

An important observation made at the PFBR site during the December 2004 tsunami was a large quantity of shells found spread on a 10 m high hillock, around 250 m from shore. This hillock was formed by excavated earth from the nuclear island area and was free of shells prior to the tsunami. We predicted that the water surge from the sea during tsunami might have transported the shells. The tsunami water surge at Fukushima reconfirmed our assessment that occasionally high surges of water will reach higher than the tsunami wave and this could travel a greater distance. As a matter of precaution, the PFBR site will provide leaktight doors and leaktight pipe and cable penetrations in the nuclear island connected building (NICB) two to three metres above finished grade level to avoid water entry into the NICB.

All the routes which could take seawater into the plant were analysed. Based on this study three routes found to be vulnerable were addressed.

The seawater intake tunnel can bring seawater into the pump house. Water may travel to various places inside the power island, through tunnels and trenches which connect the seawater pump house to other buildings. This situation may have to be encountered if any future water rise in the sea is one metre higher than the 2004 tsunami. The pipe and cable tunnel and trenches emanating from the seawater pump house will thus be sealed for watertightness.

The power island finished floor level is 2.3 m above the 2004 tsunami water rise (4.8 m). It is evident that the sea had not risen by 4.8 m; it was surge plus wave run-up which raised water to various heights (maximum 4.8 m) on the shore. In the case of sea swells of 7.2 m, seawater can enter the finished floor level of the power island through forebay and pump house doors and other openings. As a matter of extreme precaution, the seawater pump house and the forebay are planned to be surrounded by a 2.2 m high wall, to bring it to level with the finished floor level, which will prevent seawater entry unless a sea swell of more than 9.2 m occurs, which is not possible along the Kalpakkam coast.

The tsunami wall which was constructed in between the nuclear island and the sea on the beach is planned to be extended further to create a barrier between the power island and the sea. Further, the wall will be extended to a certain computed length on the sides to prevent water entry through the north and south. The tsunami wall height is planned to be raised to take care of uncertainity in our current estimate on tsunami height.

The stormwater drains are adequately designed to take care of site draining for the next 1000 years of cyclonic rainfall. In addition, the drainage system will be able to account for the water crossing the tsunami bund due to splash, if any. The stormwater drains from the plant penetrate through the tsunami wall and discharge water to sea. Non-return valves are planned to be provided in the storm water drain channels to avoid floodwater entry to the plant through the stormwater drains.

The area between the nuclear island and the tsunami wall is three metres lower than the nuclear island level. This can accumulate a large quantity of water, thus avoiding flooding, despite excessive precipitation or water coming in through splash.

Raw water reservoir breach and consequent flooding has been studied. By engineered solutions it has been ensured that the nuclear and the power island remain unaffected, even in the event of breach of the reservoir.

Two roads leading to the site and within the site to the plant are parallel to the coastline. These roads remained unaffected during the 2004 tsunami. A road is, however, planned to be constructed which runs at right angles to the shoreline and joins the existing road at the site boundary. This action is planned as a precaution; the elevated road will facilitate transportation of essential machinery, spares and supplies to the site in the case that existing approaches become unavailable.

Besides the above civil works, many more additional features were introduced in the plant. These additional features span mechanical, electrical and management aspects of the PFBR plant. This includes provision of additional diesel generators (DGs) for emergency power supply, in the case that all the existing emergency DGs become unavailable.

The PFBR has four emergency DG sets, each of 4.5 MVA capacity. Two DGs are located on the east of the nuclear island connected buildings and two on the west. While the DGs on the east are towards the sea, around 300 m from the coastline, the DGs on the west are around 450 m from coastline, in the shadow of the NICB. There is no possibility of splash of water or surge plus wave run up in the event of tsunami inundating the DGs on the west. The two DGs on the west are capable of meeting the entire design intent and ensure safety during a station black out scenario. Yet, as a precaution, it is planned to have two mobile additional DGs having the following features: 500 kVA; 415 V; 3-phase; 50 Hz; air cooled – radiator. Coolant top-up is required only after 72 hours of operation. Fuel tank capacity is for 8 hours of operation.

The 415 V loads which will be supplied by additional DGs are: 240 V AC uninterrupted power supply (UPS), 48 V DC charger, 220 V DC charger loads, 2 pony motors. There will be a biological shield cooling pump, emergency lighting (main control room, backup control room, 6.6 kV room, DGB-1 and 2, corridor from main control room to backup control room), main control room and backup control room ventilation, miscellaneous requirements (spent fuel bay instrumentation, argon and nitrogen cooling system, reactor containment building ventilation system) and power supply to the air compressors of emergency DGs.

Reactor trip on seismic event where PFBR site seismic values are:

— Operating basis earthquake: 0.078g (horizontal direction), 0.052g (vertical direction)
— Safe shutdown earthquake: 0.156g (horizontal direction), 0.104g (vertical direction)

Instrumentation has been provided in the design of the PFBR for sensing and recording seismic events. Free field sensors are planned (three) and tri-axial accelerometers (five). Free field sensors are widespread in the plant boundary; tri-axial accelerometers too are widely spread in nuclear and power islands. These are positioned in the reactor containment building (three locations) and in the service water pump house (two locations).

If the response spectrum check and the cumulative absolute velocity check are found to exceed the threshold values corresponding to the operating basis earthquake level, an annunciation will be generated in the main control room. In response to the annunciation in the control room, a field walk through will be carried out. The decision for manual shutdown of the reactor will be taken based on walk through observations and the operating basis earthquake exceedance criteria evaluated by the seismic instrumentation and the associated software. Post-Fukushima, a reactor trip on cumulative absolute velocity corresponding to a safe shutdown earthquake is planned to be introduced. The signal for trip will be generated on the basis of inputs from three free field sensors.

Surveillance testing, with adequate frequency, will be adopted for checking the functionality of the annunciation circuit for seismic trip.

Solar panels for light have independent cabling and will be installed at important locations in the plant.

The current battery bank systems are located at higher elevation and will be available during station black out conditions and during natural calamities (beyond design basis accident conditions) for four hours to cater to critical loads. The proposed additional DGs (2 × 500 kVA) can be connected appropriately to the existing battery chargers within four hours to avoid failure of the battery banks.

For the beyond design basis station blackout resulting in extended station blackout, failure of pipelines and components or small cracks in the spent fuel storage bay (SSSB) were studied to determine back fits required for the spent fuel storage bay.

Station black out, whose maximum duration is expected to be 14 hours, is considered as design basis which is a Category 4 event. Under this situation, the SSSB pool temperature rises to 42°C under normal storage and 52°C under a full core unloading situation after 14 hours. These temperatures are well within the limits. The water column available in the SSSB from the bottom is 9000 mm, against the minimum water column requirement of 8020 mm from bay bottom for shielding. The 14 hour period, with full core unloaded, provides sufficient time to take any necessary action to replenish water in the pool in the case of a small leak.

Two firewater terminals (hydrants) have been identified near to the SSSB pool to replenish the pool water. The required hose length is approximately 20 m for the first hydrant and around 30 m for the second one. In addition, bore wells with a minimum yield of 20 m^3/h are being made to provide water to replenish levels in the SSSB. Pump power supply for these bore wells will be supplied from portable DGs.

It will be ensured that pipes are permanently installed at the fuel building to facilitate quick connection for the supply of water from emergency sources to the SSSB. Also, a sufficient number of hoses (including easily connectible spools) will be kept available in the emergency equipment room and in the stores. Periodic surveys will be conducted to ensure the integrity of these hoses, which will be replaced as and when required.

Four portable pumps with diesel engines will be positioned at safe locations in the plant for emergencies. Provisions will be made to connect these pumps to make up the SSSB water level.

Additional diverse instrumentation, independent of the main plant instrumentation system will be provided for level and temperature measurements of the SSSB. Simple float type level indicators, with level markings in the pool and a suitable mechanical setup to transmit the indication to the exterior are the options available in the case of failure of the main plant instrumentation. Field measurement of temperature is also considered.

SSSB area radiation monitoring by an independent battery powered UPS having at least 4 hours capacity will be provided as a back-fit from the Fukushima review.

Flooding with seawater in the event of a tsunami is not expected, in view of the watertight shutters on all sides of the NICB, which have been introduced after the Fukushima incident.

Further, the difference in elevation between the water level maintained in the SSSB pool and the top of the SSSB wall is 2 m as opposed to the maximum surge height during a safe shutdown earthquake event of 1.1575 m. No spillage of SSSB poolwater outside the storage pool is thus envisaged.

An emergency response centre is planned to be constructed that will be well equipped with communication facilities, plant manuals and emergency operating procedures. The facility will be equipped for an extended period of stay for the emergency operating crew. The emergency response centre will be easily approachable and designed to remain unaffected for severe earthquake conditions.

4. SUPPORT SYSTEM TO OPERATOR DURING EMERGENCY

Two roads currently lead to the PFBR site. Both roads are parallel to the coast at a distance of more than 500 m from the coastline. Both roads remained intact during the 2004 tsunami. An alternative road at a higher elevation is under construction within the Kalpakkam site boundary. This road is further away from the coast. A road is planned within the plant boundary that will join the newly constructed road at a higher elevation. This proposed road will be sufficient to provide supplies and personnel to the plant during an emergency.

An emergency store equipped with essentials tools, machinery and spares is planned at a safe location to provide support to the plant during an emergency if the main store becomes unavailable. Further, clearing and cleaning machinery and emergency lighting provision will be maintained as a back-up in the emergency store.

5. MANAGEMENT ACTIONS TO COPE WITH FUTURE BEYOND DESIGN BASIS ACCIDENT SCENARIOS

The management action plans to cope with beyond design basis accident scenarios were reviewed. These reviews exhibited that a robust mechanism is in place to cope up with situations not conceived by design. The off-site emergency management plans are well developed and frequently rehearsed in India. It has been realized that Government bodies are well geared up to handle any eventuality in the public domain. The plant management's concentration, thus, has to be focused on the events and situations arising within the plant and site boundaries.

Emergency response teams have been constituted and plant managers and operators are given training to handle emergency situations. By repeated training, the psychological factors to be encountered during real emergency scenarios are addressed.

6. CONCLUDING REMARKS

The Indian Ocean tsunami of 26 December 2004 has provided enough inputs to BHAVINI to promote precautionary measures to be taken against any future tsunami. Fukushima provided further inputs to remain prepared for beyond design basis scenarios.

BHAVINI has provided engineered solutions in the PFBR to account for major tsunamis and earthquakes. Yet the preparation has not been limited to confront only the Fukushima type natural event but preparedness has extended to any future beyond design basis accident scenario which we might not have been conceived up to now.

ACKNOWLEDGEMENTS

The author acknowledges the contributions made by the designers of the Indira Gandhi Centre for Atomic Research, Kalpakkam, the team of BHAVINI (Bharatiya Nabhikiya Vidyut Nigam Limited), Kalpakkam, which has detailed the safety upgrades requirement, and the inputs received from Indian industries.

The author also acknowledges the deliberations in WANO and IAEA forums which have provided feedback on safety upgrading being done at various power plants in the wake of the Fukushima accident.

BIBLIOGRAPHY

GOVERNMENT OF INDIA ATOMIC ENERGY REGULATORY BOARD, Report Of AERB Committee to Review Safety of Indian Nuclear Power Plants Against External Events of Natural Origin", Mumbai, India (2011).

WORLD ASSOCIATION OF NUCLEAR OPERATORS, 2013 Year-End Highlights Report, London, UK (2013).

SFR SAFETY CONSIDERATIONS IN LIGHT OF THE FUKUSHIMA DAI-ICHI ACCIDENT

A. YAMAGUCHI
Osaka University,
Osaka, Japan

Abstract

The present study analyses the Japanese Government's report to the IAEA in June 2011 and the Nuclear and Industrial Safety Agency's report in March 2012 that identify lessons learned from technical and engineering viewpoints. The lessons learned are to be reflected on safety enhancement of sodium cooled fast reactors (SFRs). Safety design, accident phenomenology and accident management with respect to the Fukushima Dai-ichi accident are to be investigated through the perspective of the SFR specific features of safety. This is to ensure the SFR safety is discussed in light of the Fukushima Dai-ichi accident and lessons learned from the two reports are investigated and SFR safety considerations are extracted.

1. INTRODUCTION

Accident scenarios for sodium cooled fast reactors (SFRs) are derived from the consideration of essential safety functions, i.e. reactivity and power control, heat removal and transfer to ultimate heat sinks, to contain hazardous materials and no release of them to off-site. The accident scenarios are categorized as unprotected (without scram) and protected (with scram) scenarios according to success or failure of the first safety requirement, i.e. reactivity and power control. The unprotected scenarios are designated as anticipated transient without scram in which criticality control is lost. The protected scenarios are protected loss of core coolability (PLCC) that can be separated to two types: loss of heat sink (LOHS) and loss of reactor level (LORL). A loss of coolant accident (LOCA) scenario is improbable in a low pressure SFR system. Instead, relatively low decrease of coolant inventory in a reactor vessel, i.e. LORL is taken into account.

The earthquake off the Pacific coast of Tohoku that struck Japan at 14:46 on 11 March 2011 brought inconceivable damage to the Pacific coastal region. At magnitude 9.1, it is the largest earthquake ever recorded in Japan. The ensuing tsunamis came in a series of 7 waves that covered 561 km^2 of coast leaving 25 000 people dead or missing.

The severe accident at the Fukushima Dai-ichi nuclear power station (Dai-ichi NPS) is a LORL type that started from station blackout (SBO) and loss of ultimate heat sink. After the core coolant injection function is lost, the water level in the reactor pressure vessel (RPV) gradually decreased resulting in an uncovered core. Finally, seawater was injected with fire engines and it was taken over by pure water injection that brought the reactor and the spent fuel pool under control. Another important lesson is that the containment could not endure the high temperature and high pressure situations for a long period of several hours. Hydrogen, noble gases and volatile fission products leaked out of the pressure containment vessel (PCV).

The Japanese Government report to the IAEA in June 2011 [1] and the Japanese regulatory body (NISA) report [2] have identified the lessons learned from the technical and engineering viewpoints. We must understand the lessons thoroughly and apply them to enhancement of SFR safety. Safety design, accident phenomenology and accident management in the Dai-ichi NPS event are to be investigated through the perspective of the SFR specific features of safety. The present paper summarizes the lessons learned from the two reports and discusses SFR safety in light of the Dai-ichi NPS accident to extract SFR safety considerations from personal viewpoints. In terms of the seismic ground motion measured at the basement level of the reactor building of the Dai-ichi NPS, most of the maximum seismic ground motion observed did not exceed the design seismic ground motion except in the east/west direction for Units 2, 3 and 5. On 11 March 2011, two massive tsunamis struck the Dai-ichi NPS, the first at 15:27 (41 minutes after the earthquake struck) and the second at 15:35. The design basis tsunami water level is 3.1 m. According to an assessment (2002) based on Tsunami Assessment Methods for Nuclear Power Plants by the Japan Society of Civil Engineers, the revised tsunami water level was 5.7 m. However, the tsunami height on 11 March reached 14–15.5 m.

2. SFR SAFETY CONSIDERATIONS: JAPANESE GOVERNMENT REPORT

The Japanese Government report describes the reason that the Dai-ichi NPS accident is remarkably specialized and the lessons learned are important to be considered by the nuclear community as follows. With regard to on-site issues, it was triggered by a natural disaster; it led to a severe accident with damage to nuclear fuel, and RPVs as well as PCVs were degraded to maintain the fission product inside; and multiple units were involved at the same time. Moreover, a long term initiative is needed to settle the situation that imposes a large burden on society, such as a long term evacuation of residents in the vicinity, as well as having a major impact on industrial activities in the related regions. There are thus many aspects different from the accidents in the past, e.g. the Three Mile Island accident and the Chernobyl accident.

The lessons learned are classified into five categories. It is inevitable to carry out a fundamental review of nuclear safety measures based on the lessons. Categories 1–3 are related to the defence in depth concept, i.e. severe accident prevention, severe accident mitigation, and emergency responses. The accident at the Dai-ichi NPS was a severe accident and characterized as an event in which accident management (AM) was necessary but not successful enough. Also, off-site emergency responses were taken for resident evacuation and food and water control. Furthermore, general lessons concerning the safety infrastructure (Category 4) and safety culture (Category 5) are identified.

2.1. Lessons for severe accident prevention (Category 1)

The lessons in Category 1 are required to prevent the occurrence of severe accidents. Eight lessons in this category are: (1) safety for earthquakes and tsunamis; (2) reliability of power supplies; (3) robust cooling functions of the reactor core and the primary containment vessel; (4) robust cooling of the spent fuel pool; (5) comprehensive accident management for preventing a severe accident; (6) consideration of multiple units accident; (7) consideration of site arrangement and associated interaction; and (8) watertightness of safety related facilities.

Among those, the lessons that should be investigated prudently and in detail from the SFR safety viewpoints are discussed in the following. The earthquake was an extremely significant event that was caused by the activation of plural seismological asperities. The tsunami on the Dai-ichi NPS was as high as 15 m, which exceeded the design consideration. It suggests that the beyond design basis situations are to be considered in an appropriate manner and to the proper extent. It is noted here that the beyond design basis implies two situations: exceedance of design requirement level and occurrence of unexpected events. When we define the design requirement level of an event, the recurrence period and the magnitude of the consequence are appropriately considered so that the level is consistent with the safety goal. A tangible design based on defence in depth is essential for SFR safety considerations.

One consequence of the earthquake at Dai-ichi NPS was loss of off-site power. It is not safety grade equipment. Other damage caused by the earthquake on the safety related equipment and systems are not identified. This suggests that the current seismic design technology is highly qualified while the off-site power is vulnerable to strong earthquakes. The tsunami which followed the earthquake was devastating. The major result of the accident was SBO. The redundancy of the power supply was lost by the failure of multiple AC power systems. The requirement on redundancy is not only applied to an emergency diesel generator but to other supporting systems such as switchgear. Another insight is that the recovery procedure of AC power supply should be prepared carefully in advance. Flexibility is very important because a common cause failure by external events may affect conventional systems at the same time. In the Dai-ichi NPS event, the operators tried to use several switchgears still available after the tsunami to recover the AC power by connecting them with power vehicles but in vain. A procedure to utilize flexible measures as accident management should be determined and reviewed.

The AC power is not necessarily required for cold shutdown of the SFR because of natural circulation heat removal capability. However, the instrumentation and control (I&C) of plant parameters is important to take appropriate actions in the case of accident management. For this purpose DC power is needed and the recoverability of AC power assures the availability of the I&C capabilities. In Unit 1 of the Dai-ichi NPS, all of the power, including DC batteries, was lost. Although decay heat removal in the SFR is passive and inherently safe, many support systems and I&C require DC battery power. A possibility of sodium freezing is an example that shows the importance of I&C in preventing sodium from freezing. It is appropriate to prepare reliable I&C systems for the SFR.

The ultimate heat sink was lost in the accident because the seawater pump was submerged under seawater due to the tsunami. Also, the failure in RPV depressurization resulted in unavailability of continual operation of high pressure injection, low pressure core cooling and finally core melting. However, if an alternative heat sink were available, there would be a possibility of recovering the core cooling system. From this viewpoint, a low pressure SFR system has an essential advantage for reactor core cooling. Sudden loss of coolant is unlikely and depressurization is not necessary. Loss of ultimate heat sink resulted in failure of PCV cooling. The pressure and the temperature in the PCV went beyond the operational limitation. The situation continued for several hours and this deteriorated the PCV flange gasket and hydrogen and radioactive material leaked into the reactor building. Even if the PCV structure were still intact, every leakage possibility is to be investigated. If PCV cooling or venting worked effectively, significant release of radioactive materials would be avoided.

In the Dai-ichi NPS, loss of power resulted in loss of cooling function in the spent fuel pool. Since the heat capacity of the spent fuel pool is large enough, there was a sufficient grace period for the recovery of the function. In external fuel storage of the SFR, fresh fuels and spent fuels are stored as well. Ensuring robust cooling by introducing alternative cooling measures making the most use of passive natural circulation and atmospheric ultimate heat sink can avoid risk from the ex-vessel storage tank.

Accident management is a series of actions during the evolution of a beyond design basis accident to: (a) prevent the escalation of an event into a severe accident; (b) mitigate the consequences of a severe accident; (c) achieve a long term safe and stable state. Severe accidents are termed as accident conditions more severe than a design basis accident and involving significant core degradation [3]. Procedures to prevent an unpostulated event proceeding to a severe accident (action (a)) are included in this category. The accident management procedures prepared in advance are recovery and provision of power supplies and core and containment cooling in the Dai-ichi NPS. It should be noted that those are based on a premise that power supplies are available. The tsunami jeopardized the diesel generators as well as switchgears and DC batteries. Therefore, it was difficult to recover them within an allowable period. Probabilistic risk assessment (PRA) for external events is useful and the most use of PRA is to be made for effective accident management procedure development.

In the Dai-ichi NPS, Units 1–3 were in full power operation and Units 4–6 were in periodic shutdown. In Unit 4, the RPV inner shroud was under replacement and all the nuclear fuels in the core were transported to the spent fuel pool. Following the tsunami, all the units were in loss of ultimate heat sink and Units 1–5 were in SBO. Not only the reactors in power operation, but also those in shutdown were in a serious situation. It is noted that AC power was supplied to Unit 5 from an emergency diesel generator in Unit 5 as an accident management operation. We recognize the multiple units site has both advantages and disadvantages in terms of units' interaction. System interaction of frontline systems as well as supporting systems should be investigated so that the accident management procedures effectively work and do not affect other safety functions. Examples are flammable materials such as oil and gasoline. According to the results, the effectiveness of the physical separation strategies should be evaluated appropriately.

High radiation levels and harsh accessibility prevented the operators from taking appropriate accident management actions. The high radiation level was caused by the source terms outside the RPV, i.e. the spent fuel pool and relocation of source terms. The contaminated water in reactor buildings flowed into turbine buildings in the accident. All possibilities that impede accident management activities must be investigated in detail.

The accident was caused by the tsunami that submerged many safety related facilities, including the component cooling seawater pump, the emergency diesel generators, the switchgears, etc., impairing power supply and core and PCV cooling. The common cause may come from river flooding, heavy rain, dam break and internal flooding, as well as tsunami. Every flood source must be investigated carefully and the influences on the systems that contain sodium should be understood. The multi-barrier concept is to be applied to SFRs: (1) flood source suppression, (2) watertightness in buildings and compartments, (3) isolation and blockage of flooding routes, (4) management procedure to drain the water.

2.2. Lessons for severe accident mitigation (Category 2)

The lessons in Category 2 require response to the occurrence of severe accidents and to mitigate the consequence to prevent significant radioactive material release off-site. Seven lessons in this category are: (1) prevention of hydrogen explosion, (2) ensuring containment venting, (3) maintaining the accident management environment, (4) radiation exposure control, (5) training for severe accident management, (6) instrumentation and control to identify the status of the reactors and PCVs of multiple units, and (7) centralized control of emergency logistics.

The hydrogen explosion in Unit 1 at 15:36 on 12 March 2011 is one of the most embarrassing and influential events in the whole story of the accident. The PCV was in an inert condition with a nitrogen charge and was pressure resistant. The hydrogen explosion in the reactor building was not considered nor expected and was a surprise. It seems no precautionary measure was taken before the next explosion in Unit 3 at 11:01 on 14 March. Further, an explosion occurred at around 06:00 on 15 March 2011 in Unit 4, which had no fuel in the reactor. After the hydrogen explosions, most of the accident management activities were jeopardized, i.e. recovery of the power supply using power vehicles and cable arrangement; cooling water injection from external water sources using fire engines and hose arrangement. In addition, the workers were not sure if the operations were safe or not. It was uncertain what was going to happen next. In accident management, staff judgement and confidence in operations are important. In this context, explosions and any other possible phenomena that may influence accident management activities are to be thoroughly investigated for the SFR.

Maintenance of containment performance once a severe accident occurs is a very important function in severe accident management. In a light water reactor (LWR), the containment venting system plays an important role because the reactor core cooling generates water vapour that increases the pressure in the RPV and the PCV. The PCV vent has two functions: to reduce the internal pressure and to decrease the radioactive material concentration. Furthermore, the time duration of high temperature and pressure situations is important. It is noted that the SFR has no predominant pressurization source in the system because the operational pressure is almost atmospheric and no water is used in the system. Even so, it should be investigated extensively for the failure modes of the containment. Sodium–concrete reaction, corium–concrete reaction, sodium fire and sodium–moisture reaction are possible causes of temperature increase and pressure escalation. Containment volume and the containment cooling system and depressurization system functions will be optimized.

Several other lessons are general items concerning the accident management environment, such as habitability in the main control room in terms of radiation dose level, communication tools and lighting, and the integrity of the supporting functions necessary to continue accident response activities. Similar provisions that are to be prepared in the LWR for the fulfilment of severe accident management include measures such as radiation shielding in the control rooms and the emergency centre, the ventilation and air conditioning systems and the communication and lighting systems, without use of AC power supply for a certain period.

As to the severe accident management operations of personnel, dosimetry and atmospheric radiation monitoring were not always successful and the number of radiation protective suits and gear were not sufficient. In the SFR, control of radioactive materials as well as the sodium and its chemical composite are to be considered and managed. Adequate training and drills are essential in the accident management operations. Education on handling the sodium should be performed in advance for the workers as well as personnel in other supporting organizations, such as the self-defence forces, the police department and the fire authorities

The instrumentation of the reactor and coolant system and the PCV may not function correctly during a severe accident. In these situations, it is difficult to promptly and adequately obtain important information to identify how the accident is developing. This includes details on the sodium levels, the temperature of the reactor and the PCV, and the sources and amounts of released radioactive materials. The essential plant parameters necessary for understanding the plant status and for appropriate accident management in the SFR have to be defined according to various accident scenarios. Also, the reliability and credibility of the measurement system in severe accident situations are to be qualified.

The logistics or external support that is to be prepared in advance for severe accidents has to be investigated and identified. According to the investigation, the equipment and logistics needed for the on-site management can be defined. Specific requirements to the SFR that uses sodium as coolant need to be identified.

2.3. Lessons for emergency response (Category 3)

The lessons in Category 3 are required to respond to emergency situations upon occurrence of severe accidents and upon failure in managing the severe accident regarding preventing significant radioactive material release off-site. Seven lessons in this category are: (1) responses to combined emergencies of large scale natural disasters and a prolonged nuclear accident, (2) reinforcement of environmental monitoring, (3) establishment of a clear division of labour between relevant central and local organizations, (4) enhancement of communication relevant to the accident, (5) enhancement of responses for assistance from other countries and communication to the international community, (6) adequate identification and forecasting of the effect of released radioactive materials, (7) clear definition of widespread evacuation areas and radiological protection guidelines in a nuclear emergency.

There was difficulty in communication, providing human support, and procuring supplies from other areas as emergency responses to a significant natural disaster. In this regard the evacuation of residents took a long period of time. Most of the lessons learned concerning the emergency response are common to LWRs. Effective and swift technical advice must be provided to respond to a severe accident. However, the number of specialists in the SFR safety field is limited compared with those of the LWR. Therefore, international mutual support is very important and should be extensively strengthened. Information on the emergency should be shared within the international SFR society to enable collaborative measures for emergency response based upon sound engineering judgement. From this viewpoint, common safety design guidelines and requirements of the SFR are preferable, which would be lead by an international framework.

2.4. Lessons for reinforcement of safety infrastructure (Category 4)

The lessons in Category 4 are required to strengthen the robustness of safety infrastructure. Five lessons in this category are: (1) reinforcement of safety regulatory bodies, (2) establishment and reinforcement of the legal system, safety criteria and guidelines, (3) human resources for nuclear safety and nuclear emergency preparedness and response, (4) ensuring the independence and diversity of safety systems, (5) effective use of probabilistic safety assessment in risk management.

As to regulatory body reinforcement, it is pointed out that the Japanese regulatory body NISA is part of the Ministry of Economy, Trade and Industry, which promotes nuclear energy development and deployment. One concern is that the strict independence of the two organizations is not simple since SFRs are under development and are not fully mature. Harmonization of a high level of safety regulation and encouragement of technology development that educes the advantages and benefits, is necessary. The safety requirements for the next generation reactors are to be established in the international framework.

Safety provision requirements against the beyond design basis situations such as SBO and ULHS are currently being discussed for LWRs. On the other hand, Japanese practice on Monju safety regulation [4] has considered so-called beyond design basis events such as an anticipated transient without scram and loss of piping integrity accident in which a guillotine break of large diameter piping is postulated. An SFR has very contrasting safety features to those of LWRs. It is a low pressure system and with sudden loss of coolant. It has the capability of natural circulation decay heat removal. Sodium boiling could introduce a positive void coefficient in SFR. Therefore, the safety design guideline should be established on the basis of the safety features of the SFR under consideration.

In relation to lesson 25, the Japanese Government report concludes that "all the experts on severe accidents, nuclear safety, nuclear emergency preparedness and response, risk management and radiation protection should collaborate to address an accident by making most use of the latest and best knowledge and experience." Engineers and researchers with expertise in SFR technologies are limited. An international technical support system and development of an SFR knowledge database may be effective for the severe accident management of SFRs.

A practical and effective approach to achieve the high reliability of safety functions must be established. For the purpose, the single failure criterion, and redundancy and independence of multiple safety systems, are important safety design requirements. However, protection against common mode and/or cause failures was not appropriate in the accident. Critical multiple failure modes that may be a cliff edge of SFR safety are different from those in LWRs. For example, the SBO is not a cliff edge of the SFR since no active component is required for safe shutdown. It is important to ensure the independence and diversity of safety systems so that common cause failures can be adequately addressed and prevented and the reliability of safety functions is assured.

The last lesson in this category is active and swift utilization of PRA. The PRA is effective in identifying risk dominant accident sequences, structures, systems and components with risk importance, as well as establishing an effective accident management programme to reduce the risk. Also emphasized is a quantitative evaluation of rare and unlikely events, such as an earthquake and tsunami of exceptional magnitude. Efficient approaches to deal with the uncertainties are necessary. Experience of PRA in an SFR is not abundant. Performing PRA and accumulating experience in the effective use of risk information are required.

2.5. Lessons for permeation of safety culture (Category 5)

Category 5 consists of one lesson that concerns thoroughly instilling a safety culture. The report notes that all those involved with nuclear energy should foster a safety culture. A safety culture that governs the attitudes and behaviour in relation to the safety of all organizations and individuals concerned must be integrated into the management system [5]. Since the SFR is in a development stage and operating experience is around 400 reactor-years, it is important to have the safety culture in mind and assess all knowledge and every finding to confirm the vulnerability of a plant. It is also important to communicate with the public about SFR safety.

3. LESSONS FOR SFR SAFETY: NUCLEAR AND INDUSTRIAL REGULATORY AGENCY REPORT

3.1. Accident progression and fact finding

NISA has systematically investigated the severe accident sequence and phenomenology at each phase of the accident in detail. Figure 1 shows the progression of the Dai-ichi NPS accident. Based on the data and findings, NISA extracted technical knowledge of the incident and formulated the necessary countermeasures for protection in the event of a severe accident scenario of this kind. The results are published in a report that identifies 30 technical lessons from viewpoint of safety design and accident management. In this section, the lessons are summarized and insights applicable to SFRs are discussed.

Six transmission lines out of seven for off-site power were available (one in outage for maintenance). The circuit breakers and disconnectors were damaged, and the collapse of an embankment caused a power transmission tower to fall over. All of this resulted in a loss of off-site power. Units 1–3 automatically shut down after the earthquake; emergency diesel generators automatically started supplying power. The reactor core isolation cooling system and the isolation condenser started operation and other emergency core cooling systems were ready.

The tsunami that followed caused the SBO. The tsunami submerged the emergency diesel generator, AC power supply equipment (high voltage switchgear, power centre, etc.). These were unrecoverable. Furthermore, the tsunami also flooded the seawater pumps used to cool the reactors of all units. The residual heat removal systems and auxiliary cooling systems for supporting equipment operability could not be used anymore; it is loss of ultimate heat sink. In addition, in Units 1, 2 and 4, the tsunami disabled the DC power supply.

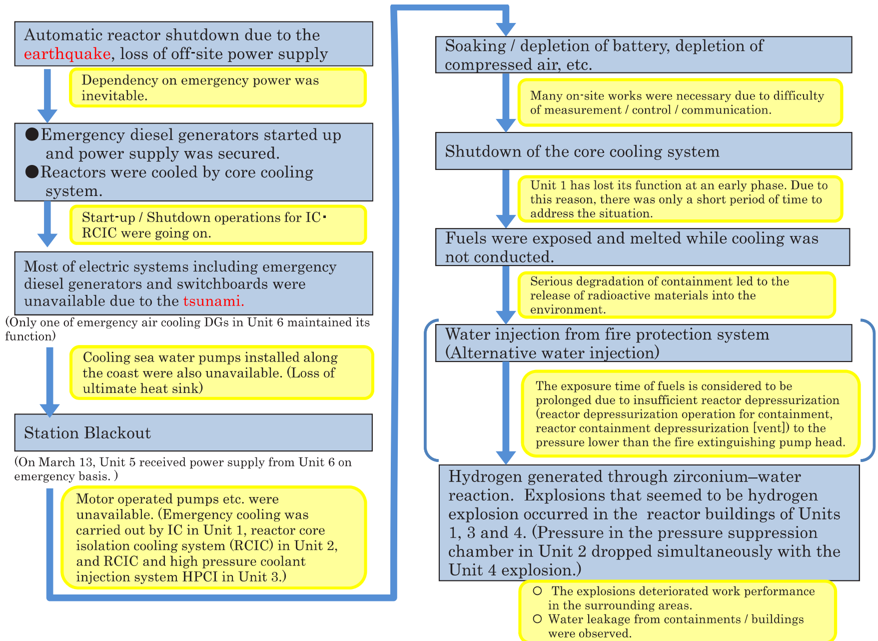


FIG. 1. Progression of severe accident in the Fukushima Dai-ichi NPS [2].

Therefore, the instrumentation system was not available, leaving the plant without control. Monitoring systems were unavailable and the motor operated valves did not work. Although the DC power supply was intact in Unit 3, the batteries are depleted sooner or later. The situations in Units 1–4 were complete loss of power.

The total loss of power resulted in failure in core cooling. The loss of water level in the core led to uncovery of the fuel assemblies and core meltdown. Hydrogen was generated due to the fuel cladding material, zirconium and water vapour reaction. The hydrogen and volatile radioactive materials leaked into the reactor buildings through the PCV resulting in a hydrogen explosion in the reactor buildings of Units 1, 3 and 4. They contaminated the area around and accident management became very difficult and was supposed to mitigate the consequence and prevent massive release of radioactive materials.

The scope of the NISA report is based on the accident sequence. It focuses on the off-site power supply (electrical substation, switchyard, etc.), on-site emergency power supply, cooling system (core cooling systems, component cooling water systems, etc.), containment system (PCV, venting, etc.), command and communication system, and I&C system. The period of the accident ranges from the occurrence of the earthquake to the release of the radioactive materials into the environment. The technical knowledge and some of the countermeasures are extracted by a bottom-up approach. In other words, some countermeasures are designed so that only the specific phenomenon and incident in the accident are prevented. The effectiveness of provisions are to be evaluated so that general external events are covered and they do not influence other safety functions.

3.2. Thirty technical lessons learned

As shown in Figure 2, 30 technical lessons are extracted based the accident sequence and the facts observed at present. According to the sequence of the accident, 30 lessons are categorized in: (1) off-site power supply system, (2) on-site power system, (3) cooling system, (4) containment system and (5) command, communication, I&C system.

The reliability of the off-site power needs to be improved by implementing regulations that apply to external power supply and grid and electrical substations directly connected to nuclear power stations. The external power supply failure may include damage to on-site and off-site equipment. Some of these are non-safety grade. It took a long time to recover the off-site power supply

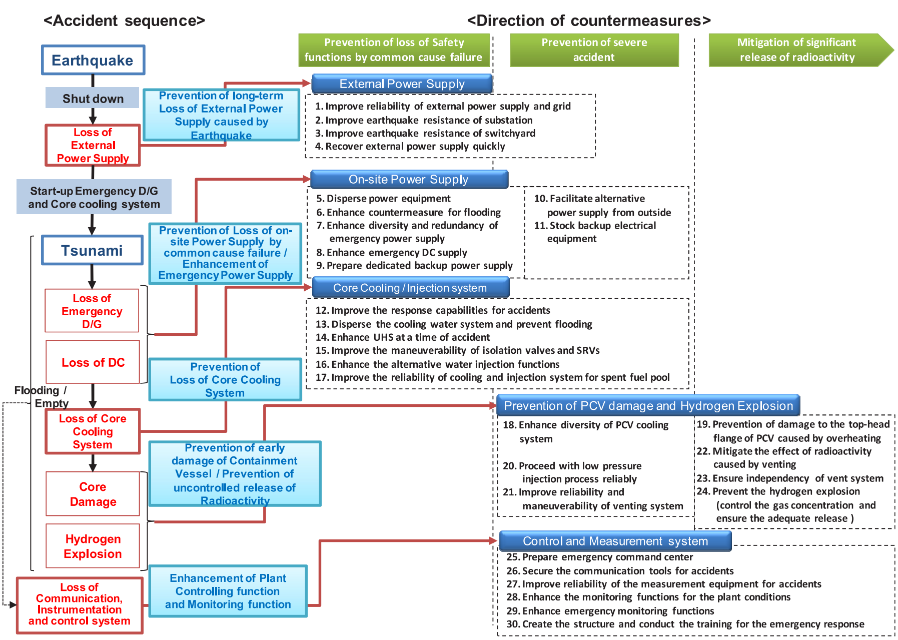


FIG. 2. Selected countermeasures (30 technical lessons) against the Dai-ichi NPS accident scenario [2].

because damaged electrical equipment could not be replaced in a short time. It is necessary to recover the off-site power supply within a certain period to reduce the risk of severe accidents.

In order to prevent an occurrence of an SBO caused by common cause failures, enhancement of robustness against flooding is required by the physical separation of the redundant on-site power supply system, the watertightness of buildings, etc. For the emergency diesel generators, ensuring the reliability of component cooling methods is necessary. With regard to DC power supply reliability, it is necessary to maintain the function for long periods of time and to install alternative charging systems for DC batteries. For the prompt recovery of the power supply systems, it is necessary to facilitate an electrical power supply from outside and to store spare parts related to the electrical equipment.

For making proper judgements in the accident management operations, it is necessary to clarify the criteria for priority selection. Provision of appropriate hardware (I&C, dosimeters, masks, etc.) and software (operational procedures) is very important to achieve successful accident management. In order to prevent the loss of cooling function by common cause failures, it is necessary to adopt anti-flooding capabilities by having watertight buildings, physical separation of locations and diversification of the ultimate heat sink. It is recognized that reactor core cooling is especially important. In order to ensure the water injection function, it is necessary to establish a compulsory operation rule. Alternative water injection provision should be strengthened by improving the operation reliability, diversification of the isolation valve control, preparation of high pressure injection pump, and installation of water inlets outside of the buildings. In order to improve the reliability of the cooling and water supply functions of the spent fuel pool, it is important to secure redundancy and diversification of the functions. It is also effective to install air cooling systems and promote the decentralization of storage by adopting dry storage, so that sufficient extra time can be secured until the corresponding cooling is required.

To prevent excessive pressure and temperature within the PCV in the event of an SBO, diversity of heat removal functions are required, including a PCV spray function and residual heat removal system that do not rely on AC power. Taking into account damage to the water intake pumps and to other facilities following this accident, diversity of PCV heat removal functions are required. The venting piping must be separated from the standby gas treatment system, and the use of common vent exhaust stacks between reactors must be eliminated, to ensure that hydrogen gas does not backflow via the standby gas treatment system or vent piping of other reactors into the reactor building. Clear procedures are required for successfully switching over to alternative low pressure water injection systems. Facilities to effectively remove radioactive materials (filters) must also be added, and manoeuverability of the venting system must be improved. To prevent hydrogen explosions in reactor buildings, the concentration of hydrogen must be properly controlled and hydrogen gas must be properly vented.

To improve the reliability of command and communication facilities, it is necessary to secure and develop an emergency command system and steadily implement measures to ensure power supply availability to maintain the functions of the major communication lines between emergency response headquarters and other facilities. Also, to take immediate and appropriate actions at the emergency response headquarters and related agencies, it is necessary to establish a highly reliable information sharing system. To have accurate information of plant conditions, it is necessary to secure the reliability of the detection devices at the time of an accident and strengthen the plant monitoring function. It is necessary to maintain manuals, design drawings and other necessary information and documents to use for emergency responses. Adequate and sufficient human resources must be available and be trained for emergency responses in various situations, even at night or in inclement weather situations.

4. DISCUSSIONS AND SUMMARY

As discussed above, the accident indicates many lessons with regard to severe accident prevention and accident management as well as emergency responses. Furthermore, several underlying cross-cutting issues such as safety infrastructure are pointed out. It is important to reflect on the lessons to SFR safety.

(1) *PRA and safety margins study for natural disasters and unlikely events:* It is important to prepare for two types of unexpected events — exceedance of design requirement level and occurrence of out-of-consideration or unlikely events. The safety margin study and PRA need to be performed and effectively utilized to identify the cliff edge, if any, and pragmatic accident management procedures adopted to prepare for unknowns. Although past severe accident scenarios have been considered in PRA, they were not reflected in safety enhancement activities. PRA and margin study experience for SFRs are limited. More resources are to be used for the safety margins and risk assessment. The assessment must be plant specific.

(2) *Passive safety features:* The SFR system has natural decay heat removal capability and large temperature margins to sodium boiling. It implies the passive heat sink to the atmosphere is available and no emergency coolant injection is required. Grace time for recovery actions is much longer than for LWRs (more than 10 hours). They are significant advantages to LWRs, whose decay heat removal capability relies on the sensible heat

of water rather than the latent heat. On the other hand, maintenance of sodium inventory is required because of the difficulty in pouring sodium in the primary system. Only if the reactor core is submerged under sodium can severe core damage be prevented. According to the characteristics of the SFR, alternative safety function, mobile equipment and emergency procedures are to be prepared for, which may be different from those for LWRs.

(3) *Containment function:* Since SFRs are low pressure systems, the safety requirement on the containment function may be different from LWRs. The only pressure source is the sodium fire inside the containment that results in elevated temperature and pressure. The compartments that involve the primary system could be filled with nitrogen. It will effectively suppress the sodium fire inside the containment vessel. In severe accidents, a certain mechanical energy release can cause sodium ejection into the air on the operational floor. The ejection of the sodium is a threat to the containment integrity. The trade-off of the containment cooling capability and the thermohydraulic response and load on the containment should be considered when defining the containment requirement.

(4) *Ex-vessel phenomena and loss of reactor level:* Another consideration on the containment is a loss of coolant situation with multiple failures of primary coolant boundaries. Although a multiple primary structure failure is very rare, core uncovery could occur in some situations. In a PRA, the scenario is to be studied and possible severe accident management procedures such as ex-vessel long term cooling will be investigated to prevent containment failure and significant radioactive material release.

(5) *Accident management:* Active operations are not necessary for the accident management of the SFR and the power supply is less important. However, the accessibility to the safety systems and visibility of plant situations are inferior to LWRs. Although the reliance on the power supply is less important, reliability and mission time of DC power and I&C systems should be investigated. Possibilities and limitations of I&C system malfunction are to be investigated to ensure the accident management operations. As to the accident management operations, physical separation of safety systems and accident management equipment is very important to prevent common cause/mode failures. The possibility of adverse interactions of facilities should be identified and avoided.

(6) *Resources for severe accident management:* Lastly, the SFR experience and number of specialists are not abundant at present. From the plant system safety viewpoint, PRA and safety margin analysis are important. In the severe accident management and emergency response, knowledge, expert judgement, accident delineation capabilities and external support

are indispensable. The most practical and effective way is though well-organized international collaboration. The safety design guideline should be based on common safety criteria. Engineers and specialists should be ready to provide mutual support under extreme conditions. Resources to mitigate the severe accident consequenced are to be shared so that the public and the environment are defended robustly.

REFERENCES

[1] GOVERNMENT OF JAPAN NUCLEAR EMERGENCY RESPONSE HEADQUARTERS, The Accident at TEPCO's Fukushima Nuclear Power Stations, Report of Japanese Government to the IAEA Ministerial Conference on Nuclear Safety, June 2011.

[2] NUCLEAR AND INDUSTRIAL SAFETY AGENCY, Technical Knowledge of the Accident at Fukushima Dai-ichi Nuclear Power Station of Tokyo Electric Power Co., Inc., (Provisonal Translation), March 2012.

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Severe Accident Management Programmes for Nuclear Power Plants, IAEA Safety Standards, Safety Guide, No. NS-G-2.15, IAEA, Vienna (2009).

[4] THE NUCLEAR SAFETY COMMISSION OF JAPAN, NSCRG: D-FR-I.01 — Philosophy in Safety Evaluation of Fast Breeder Reactors, Japan (1980).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Fundamental Safety Principles, IAEA SAFETY STANDARDS SERIES No. SF-1, IAEA, Vienna (2006).

R&D CHALLENGES FOR SFR DESIGN AND SAFETY ANALYSIS: OPPORTUNITIES FOR INTERNATIONAL COOPERATION

N. DEVICTOR
Commissariat à l'énergie atomique et aux énergies alternatives,
Gif-sur-Yvette, France

Abstract

The paper summarizes the R&D in the safety domain in support of sodium cooled fast reactor (SFR) design and safety analysis. Examples are provided, in particular in the fields of reactivity and decay heat removal control, severe accident analysis, in-service inspection and repair, and chemical risks. It is highlighted that these activities are relevant for international cooperation, especially benchmarks and sharing of experimental facilities. Different frameworks are available for cooperation, such as the Generation IV International Forum, the IAEA (in particular through its coordinated research projects), the European Commission Framework Programme and also bilateral cooperation.

1. INTRODUCTION

The objective of the paper is to summarize the R&D in the safety field in support of sodium cooled fast reactor (SFR) design and safety analysis. Examples are provided, in particular in the domain of control of the reactivity function and the decay heat removal function, severe accident analysis, in-service inspection and repair, and chemical risks. Opportunities for international cooperation are highlighted.

2. SYNTHESIS OF THE R&D PROGRAMME IN SUPPORT OF THE DEVELOPMENT OF SFRS

The considerable experience gained with SFRs in several countries shows their favourable characteristics, as for instance [1, 2]:

- SFRs are easy to operate, owing to the absence of primary coolant pressurization, the high thermal inertia provided by sodium, and a control of neutron flux simply by control rods (no xenon effect, no need of soluble neutron poison).

- SFRs provide high thermal efficiency.
- SFR's level of radiological protection is higher than for light water reactors.
- Prevention of core degradation benefits from the large thermal inertia and a large sodium boiling margin, and the ability to operate with natural convection, as experienced on the Phénix and SuperPhénix reactors, for instance.
- The heat sink of the decay heat removal system could be diversified by using air.

The operation feedback also highlights some ways for improvement of the next generation of SFRs. Figure 1 summarizes these, for different topics, e.g. safety, operation, performance.

In the next section, examples, but not a comprehensive list, are given, particularly on the challenges related to safety and materials.

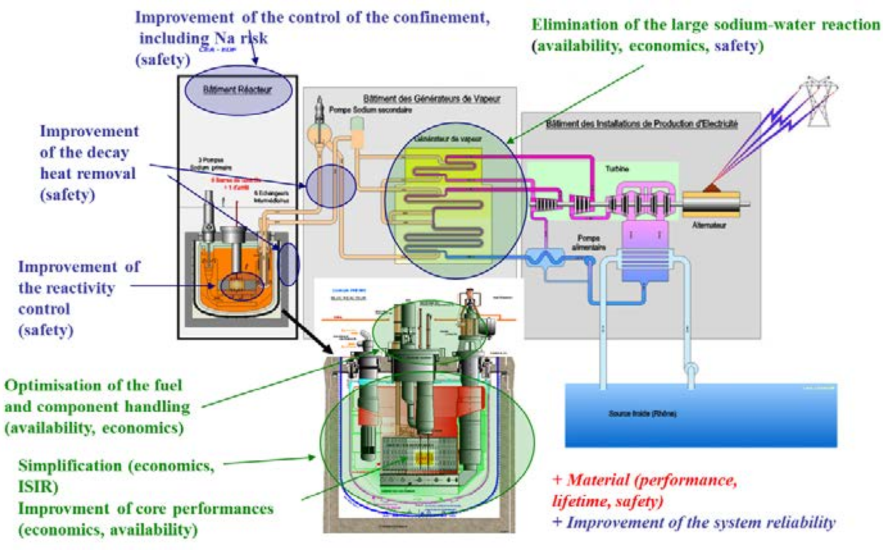


FIG. 1. Areas for improvement for future SFRs.

3. EXAMPLES OF AXES OF R&D RELATED TO THE REACTIVITY CONTROL

Trends in the core design are to enhance the natural behaviour in the case of no actuation of active shutdown systems. This has been reinforced after the Fukushima-Daichi accident. Prevention by natural reactor behaviour, in particular by core feedbacks, of any entrance in a severe accident scenario in the case of transients without scram, is considered in the ASTRID project — ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) is a project for an industrial demonstrator of the SFR led by the Commissariat à l'énergie atomique et aux énergies alternatives (CEA) [3]. Once this risk is overcome, return to a safe state (i.e. zero fission power at low temperature) is achieved over the long term thanks to a complementary safety device if needed [4].

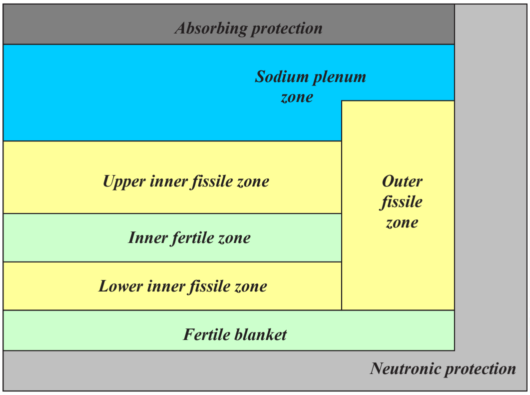
For ASTRID, specific design criteria have been taken into account:

— A natural behaviour favourable for transients of unprotected loss of flow and loss of heat sink, with the objective if avoiding sodium boiling;
— A minimization of a sodium void effect with the objective of attaining a negative sodium void worth or a value near 0;
— A natural behaviour favourable for a complete control rod withdrawal (without detection), with the objective of no fuel fusion with very high confidence.

A core with a low sodium void worth, called the CFV core, accordingly has been proposed for ASTRID (Fig. 2) [5].

Such a core presents a different behaviour compared to an optimized homogeneous core such as SFRv2 [6], as described in this paper [5].

Qualification of such an innovative core requires experimental programmes that could be suitable for cooperation. For instance, the CFV core falls outside the scope of existing experimental qualification when considering the neutron spectrum, given their different characteristics compared with the previous core designs (such as Superphénix or the European Fast Reactor project) in terms of volume fractions of fuel and sodium. Moreover, the influence of the combination of geometrical characteristics of the CFV core should be verified. The CEA proposes an experimental programme, called GENESIS, in MASURCA mock-up, and a first phase of the GENESIS programme is starting in BFS in partnership with IPPE. To complement the experimental programme, benchmark analyses are necessary in support of the verification and validation of codes and in order to reduce bias. The CEA and US Department of Energy (DOE) performed such benchmarking at the end of 2012 [7].



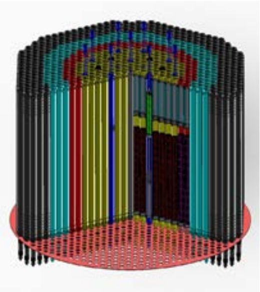


FIG. 2. CFV core.

Figure 3 explains two R&D axes in order to improve confidence in the transient analysis:

- Understanding of double phase thermohydraulics that could lead to stable sodium boiling in sodium plenum of subassemblies. Such R&D could lead to the development of specific experimental facilities and is an opportunity for benchmarking.
- Development of material laws on fuel structures for high temperature [8].

This last R&D topic needs an experimental programme under several conditions (temperature, transients, etc.) on clad and hexcan materials. As for the first ASTRID' cores, material already used in Phénix will be used (for instance AIM1 for clad and EM10 for hexcan); the exploitation of available irradiated material at Phénix is planned [9].

In the case of dynamic load on the core, due, for example, to a seismic hazard, sodium fluid provides a resistive action. A better understanding of the phenomenon could permit a better understanding of the margin in dynamic solicitation. Experiments on fluid structure interactions are ongoing, for instance in a partnership between the CEA and Indira Gandhi Centre for Atomic Research (IGCAR).

Other important R&D activities with regard to reactivity control include the development of the instrumentation for core monitoring. Requirements mainly involve diversifying the means of protection and improving instrumentation performance in terms of responsiveness and sensitivity [10, 11]. Several organizations are developing similar devices, e.g. fission chamber, flowmeter

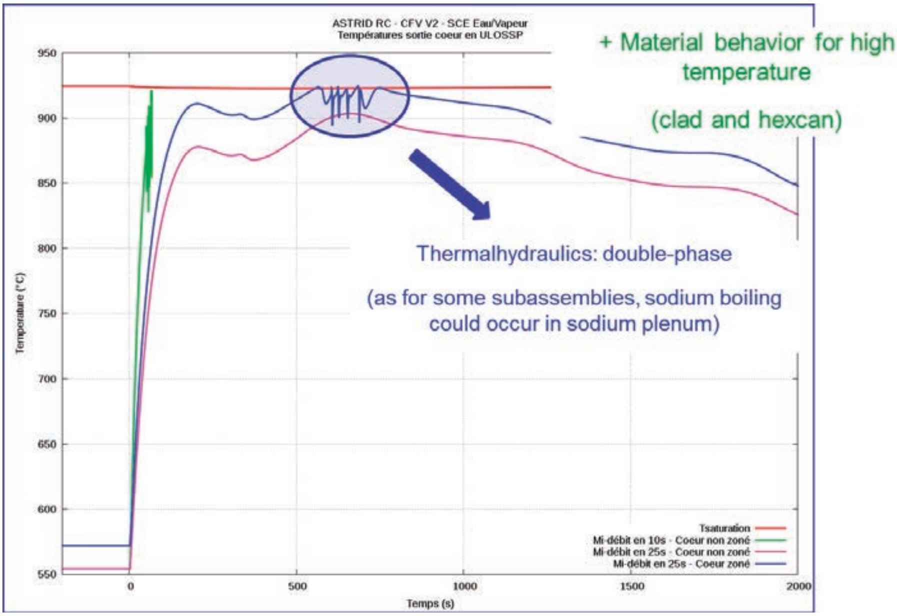


FIG. 3. R&D axis in support of transient analysis.

and ultrasonic devices. Their qualification could make use of facilities or experimental reactors of partners.

An important line of R&D, strengthened after the Fukushima-Daichi accident, concerns the post-accident instrumentation to enable monitoring the evolution of the potentially damaged core.

4. EXAMPLES OF R&D AXES RELATED TO SEVERE ACCIDENTS

Figure 4 summarizes the different steps of R&D in the field of severe accidents [12].

Given the phenomena involved during a severe accident and their interaction, the representativeness of experiences is limited by nature. Consequently, the analysis is based on mechanistic computer codes, validated on experiments simulating a portion of the scenario or separate phenomena. Sensitivity and uncertainty studies are conducted to evaluate the robustness of the analysis.

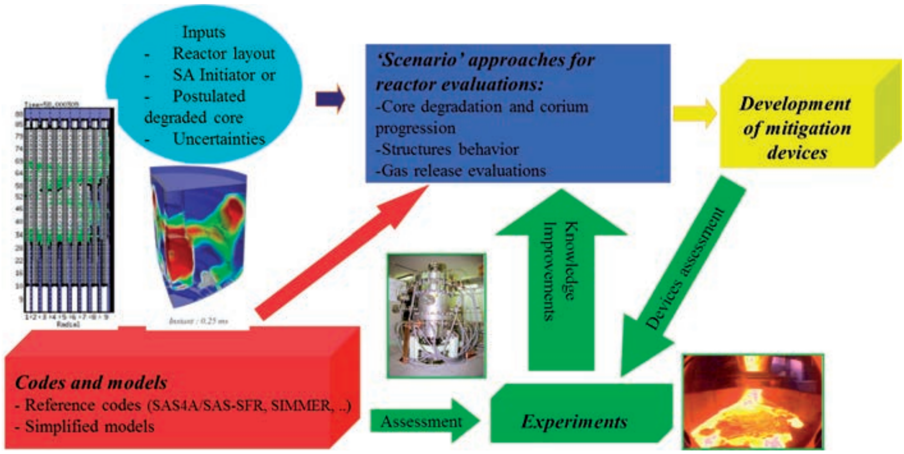


FIG. 4. R&D strategy for severe accidents.

The domain of the severe accident analysis leads to the development of long term cooperation between the different R&D organizations, designers and utilities, for the development of codes and the realization of experimental programmes. The calculation codes SIMMER and SAS4A and the CABRI and SCARABEE programmes could be quoted. Currently, experimental facilities are open to an international programme, as for instance PLINIUS Prototypic corium platform (www.plinius.eu) or IGR reactor at the National Nuclear Centre of Kazakhstan (Fig. 5), in which Japan Atomic Energy Agency (JAEA) is conducting the EAGLE programme with the participation of the CEA in support of the development of mitigation devices.

Owing to the lack of knowledge on fuel–coolant interactions and in the assessment of fission product releases from fuel elements in severe accident conditions, new experimental programmes are considered that will require the development of new facilities.

For fuel–coolant interaction, IGCAR is developing the SOFI installation. The paper [13] provides information on the CEA project to adapt the PLINIUS platform for experiments with sodium, with the building of the FOURNAISE facility (Facility for Observation of URanium oxide — NA Interaction and Safety Experiments). It should be designed for a large quantity of UO_2, with the objective of extending the database with saturated sodium, for various corium discharge flowrates (up to 300 kg), and analysing debris formation and its transport in a tube.

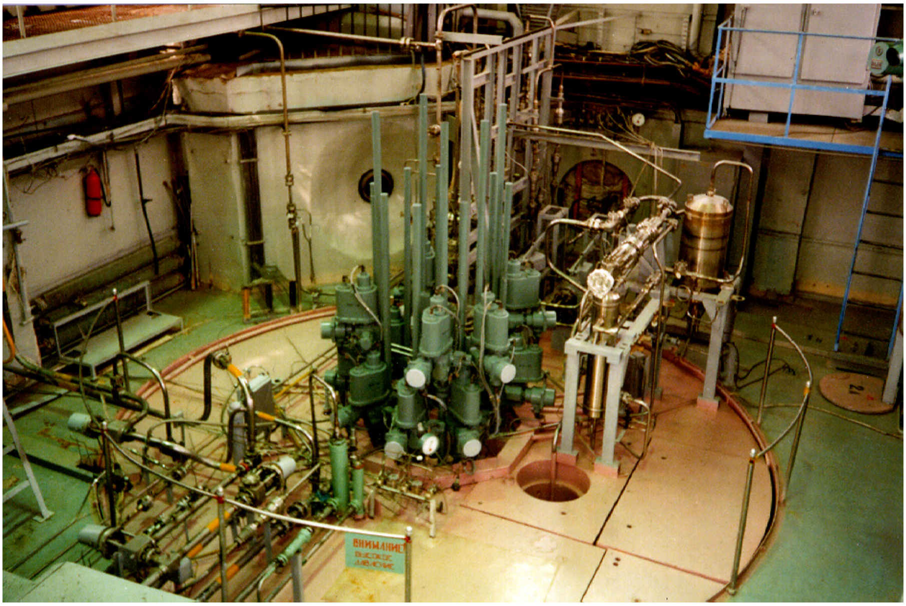


FIG. 5. View of the IGR reactor (courtesy NNC).

This R&D should be done in accordance with the design of the core catcher for SFRs. Several options are currently considered, for instance for the ASTRID reactor: internal to main vessel, or external to the main vessel with 2 options, one located outside the safety vessel, and one located in-between the primary and safety vessels (inter-vessel core catcher). To support the core catcher design, a large R&D programme is under way regarding the melt progression (characteristics of the corium arriving on the core catcher), protective materials to mitigate the thermomechanical loads of the core catcher, corium behaviour and its cooling on the core catcher, which are all being supported by experimental programmes, including the long term behaviour of materials [14].

5. EXAMPLES OF R&D AXES RELATED TO DECAY HEAT REMOVAL

One of the SFR challenges is to demonstrate the long term performance of decay heat removal system in any situation, including in the case of severe accidents (for instance when the corium is on the core catcher). As an objective, total and definitive loss of the 'heat removal' function has to be practically eliminated. In order to reach such an objective, R&D is ongoing on the minimization of common cause and the development of probabilistic safety analysis models suitable for SFRs.

For the minimization of common cause, development of efficient systems through the structure of reactor vessels is a strong R&D axis, particularly the comparison of fluid performances and the assessment of the evolution of the emissivity of structural materials.

Owing to SFR characteristics (e.g. large plena), assessment of the decay heat removal performance by system code is required to be supported by computational fluid dynamics codes. One trend is to develop computational schemes that couple computational fluid dynamics codes and system codes, as shown in Fig. 6, and simulate an unprotected loss of flow. R&D should be done in order to validate such computational schemes, and experiments as thus performed on Phénix during the end of life test are very valuable for the qualification [15]. International benchmarks are very useful in order to better understand bias and uncertainty sources [16].

Another way is the development of a dedicated experimental loop for the verification of decay heat removal performance. Well known examples are the STELLA-1 loop builds by KAERI (Fig. 7), and the ATHENA loop proposed by the JAEA.

6. OTHER R&D TOPICS

The use of sodium as coolant implies consideration of the chemical risks due to sodium fire and reactivity with water or air.

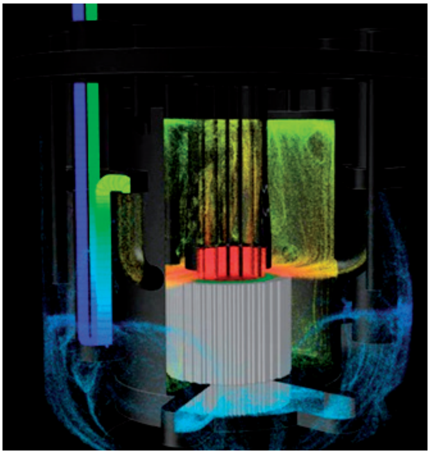


FIG. 6. Example of TrioU — CATHARE 2 calculation on a ULOF.

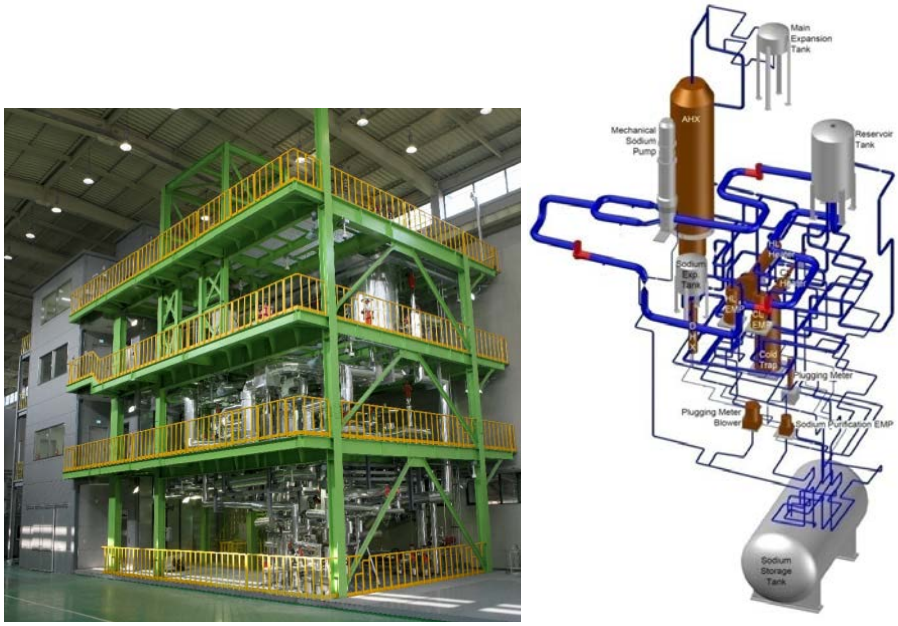


FIG. 7. STELLA-1 loop (courtesy KAERI).

R&D is ongoing with the objective to better understand the kinetics of sodium aerosols, in support of safety margin assessment (Fig. 8). An experimental programme could be useful in support of this new assessment. R&D results will also be used in the validation of codes used for safety analysis, e.g. FEUMIX, PULSAR and CONTAIN.

The objective of mastering the potential risk from a sodium–water reaction, in the case of a steam–water energy conversion system, leads to different lines of R&D [17]:

— Study of design options based on modular steam generators;
— Completion of some material databases in support of mechanical behaviour modelling and wastage modelling for new steam generator designs, for instance, 9Cr straight tubes, inverted tubes, etc., by realization of an experimental programme on facilities such as SWAT1R (JAEA) or SOWART (IGCAR) or SQUAT (CEA);
— Identification and assessment of sodium–water–air reaction accident scenarios in steam generator buildings [18].

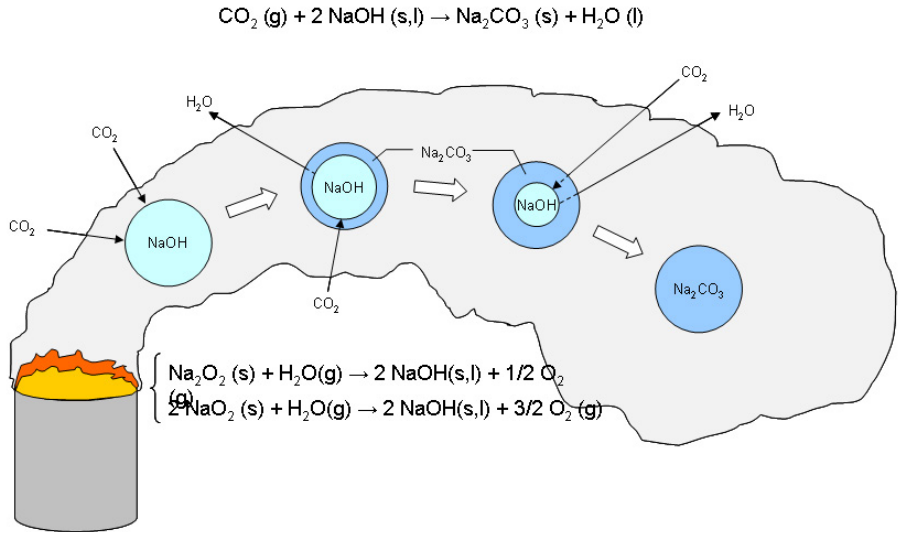


FIG. 8. Sodium aerosols.

An innovative line of R&D is to replace the steam–water energy conversion system by a gas energy conversion system based on a Brayton cycle. The paper [19] describes the current system studied as an option for ASTRID, based on pure nitrogen at 180 bars, and discusses the general assessment in terms of their performance, safety approach, operability and balance of plant.

Another main issue for R&D is the improvement of in-service inspection and repair capabilities. The periodic examination and repair are examined through the following main R&D axes [20]:

— Improvement of the primary system conceptual design in order to ease periodic examination and repair;
— Development of inspection techniques (periodic inspection tools and associated simulation);
— Development of robotics;
— Development and validation of repair processes.

Key milestones of this R&D programme are, particularly, the validation for ultrasonic simulation, the validation of ultrasonic transducers (ultrasonic telemetry and non-destructive examination sensors) under sodium conditions at 200°C (see for instance MULTIREFLECTOR mock-up in Fig. 9), the definition of key components of the robotic equipment for operation in sodium at 200°C,

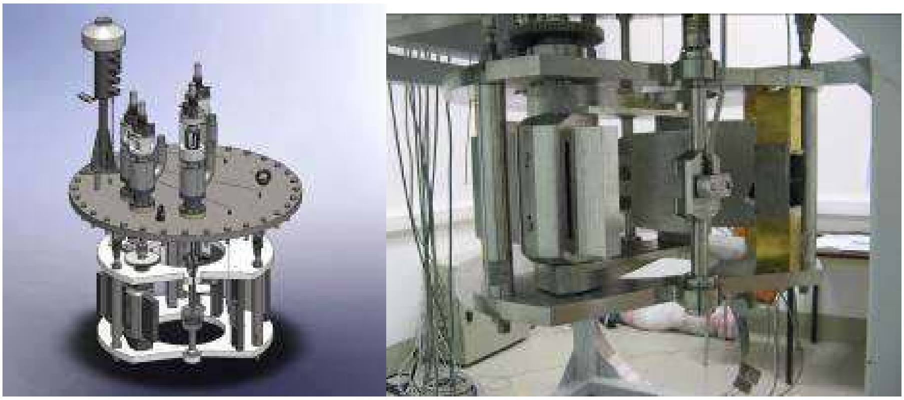


FIG. 9. MULTIREFLECTOR mock-up devoted to in-sodium ultrasonic telemetry study.

and the preliminary validation of laser repair processes and techniques (sodium sweeping, structure machining and welding).

That R&D axis is suitable for international collaboration for sharing technological development, validation of computer codes, or investment in experimental facilities.

7. CONCLUSION

The paper presents examples of R&D challenges related to the reactivity and decay heat removal control, severe accidents, in-service inspection and repair and chemical risks. For any domain, R&D activities include modelling, codes development and their verification and validation process, with the support of experimental programmes. The success in the R&D will help the safety case and the acceptability of SFR.

Some of these activities are relevant for international cooperation, especially benchmarks and sharing of experimental facilities. It could benefit from recent catalogues of experimental facilities (already operational or in a project), for example from the task group on Advanced Reactor Experimental Facilities of the OECD/NEA [21] and the European project ADRIANA (ADvanced Reactor Initiative And Network Arrangement [22].

Different frameworks are available for cooperation, e.g. the Generation IV International Forum, the IAEA (in particular through its coordinated research projects), the European Commission Framework Programme and also bilateral cooperation.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Proc. Int. Conf. Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), Kyoto, Japan, 2009, IAEA, Vienna (2012).

[2] MARTIN, L., et al., "Phénix: 35 years operation 1974-2009", Revue Générale du Nucléaire, Janvier-Février (2009).

[3] LE COZ, P., et al., "The ASTRID project: Status and future prospects", paper presented at Int. Conf. Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[4] LO PINTO, P., et al., "Safety orientations during ASTRID conceptual design phase", paper presented at Int. Conf. Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[5] CHENAUD, M-S., et al., "Status of the ASTRID core at the end of the pre-conceptual design phase 1", paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[6] SCIORA, P., et al., "A break even oxide fuel core for an innovative French sodium-cooled Fast Reactor: Neutronic studies results", Proc. Int. Conf. on Fast Reactors and Related Fuel Cycles: Challenges and Opportunities (FR09), Kyoto, Japan, 2009, IAEA, Vienna (2012).

[7] SOFU, T., et al., these Proceedings.

[8] LE FLEM, M., et al., "French R&D materials for the core components of Sodium Fast Reactors", paper presented at Int. Conf. Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[9] MUNOZ, I., et al., "Recovery of materials from PHENIX to support the qualification of ASTRID design options", paper presented at Int. Conf. on Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[10] JEANNOT, J.-Ph., et al., "R&D program for core instrumentation improvements devoted for French Sodium Fast Reactors", paper presented at 19th Int. Conf. on Capture and Gamma-ray Methods and their Applications (ANIMMA), Ghent, Belgium, 2011.

[11] JADOT, F., et al., "Instrumentation in the ASTRID project: Main stakes and feasible solutions", paper presented at Int. Conf. Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[12] DEVICTOR, N., et al., "R&D programmes related to severe accident in SFR", Proc. Int. Workshop Prevention and Mitigation of Severe Accidents in Sodium-Cooled Fast Reactors, Tsuruga, Japan, 2012.

[13] RODRIGUEZ, G., et al., "Development of experimental facility platform in support of the ASTRID program", paper presented at Int. Conf. Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[14] SERRE, F., et al., "R&D and design studies for the ASTRID core-catcher ", paper presented at Int. Conf. Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[15] FONTAINE, B., et al., "Recent analyses of PHENIX End of Life Tests and perspectives", paper presented at Int. Conf. Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[16] MONTI, S., et al., "Overall results of and lessons learned from the IAEA CRP on Sodium Natural Circulation Test Performed during the PHENIX End-of-Life Experiments ", Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP), Chicago, IL, 2012, American Nuclear Society, LaGrange Park, IL (2012).

[17] SAEZ, M., et al., "Sodium-water reaction approach and mastering for ASTRID steam generator design", paper presented at Int. Conf. Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[18] DAUDIN, K., et al., "Approach to the identification and evaluation of Sodium-Water-Air Reactions (SWAR) accidental scenarii in Steam Generator buildings", paper presented at Int. Conf. Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[19] LAFFONT, G., et al., "ASTRID power conversion system: Assessment on steam and gas options", paper presented at Int. Conf. Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[20] BAQUE, F., et al., "ASTRID In Service Inspection and Repair: review of R&D program and associated results", paper presented at Int. Conf. Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios (FR13), Paris, 2013.

[21] OECD NUCLEAR ENERGY AGENCY, Experimental Facilities for Sodium Fast Reactors Safety Studies, Task group on Advanced Reactor Experimental Facilities (TAREF), OECD Publishing, Paris (2011).

[22] LATGÉ, C. et al., "ADRIANA Project: identification of research infrastructure for the SFR, within the frame of European Industrial Initiative for Sustainable Nuclear Fission", Proc. Int. Congress on Advances in Nuclear Power Plants (ICAPP), Chicago, IL, 2012, American Nuclear Society, LaGrange Park, IL (2012).

SODIUM FAST REACTOR SAFETY AND LICENSING RESEARCH PLAN

M. DENMAN*, T. SOFU**, R. BARI***, G. FLANAGON+,
J. LACHANCE*, R. WIGELAND++

* Sandia National Laboratories, Albuquerque

** Argonne National Laboratory, Argonne

*** Brookhaven National Laboratory, Upton

+ Oak Ridge National Laboratory, Oak Ridge

++ Idaho National Laboratory, Idaho Falls

United States of America

Abstract

This paper summarizes potential research priorities for the US Department of Energy (DOE) with the intent of improving the licensability of the sodium cooled fast reactor (SFR). In support of this project, five panels were tasked with identifying potential safety related gaps in the available information, data and models needed to support the licensing of an SFR. The areas examined were sodium technology; accident sequences and initiators; source term characterization, codes and methods; and fuels and materials. It is the intent of this paper to utilize a structured and transparent process that incorporates feedback from all interested stakeholders to suggest future funding priorities for SFR research and development. While numerous gaps were identified, two cross-cutting gaps related to knowledge preservation were agreed upon by all panels and should be addressed in the near future. The first gap is a need to re-evaluate the current procedures for removing the applied technology access control designation from old documents. The second cross-cutting gap is the need for a robust knowledge management and preservation system in all SFR research areas. Closure of these and the other identified gaps will require both a reprioritization of funding within DOE as well as a re-evaluation of existing bureaucratic procedures within the DOE associated with applied technology and knowledge management.

1. INTRODUCTION

1.1. Motivation

The US Department of Energy (DOE) is currently funding various research initiatives to support future fast reactor deployment. In addition to recording new technical achievements, the ability to support a US industrial partner's licensing effort may potentially influence future DOE funded research areas. Sodium cooled fast reactors (SFRs) have long been both studied and operated by the DOE and its predecessor, the Atomic Energy Commission. By the early 1990s, the DOE complex operated a wide range of fast reactor related experimental facilities. The DOE national laboratories supported fast reactor research with both experimental facilities and predictive model development across all relevant areas pertaining to SFR deployment until the programme was suddenly closed in 1994.

In 1994, policy changes in the US Government abruptly cancelled nearly all of the fast reactor research in the United States of America. The abrupt end to fast reactor research resulted in many programmes ending prematurely, with no funding available to properly preserve intermediate or final project results. Currently, raw data tapes and paper records are stored in uncontrolled environments where they are not protected from deterioration or unintentional destruction. Even if the data tapes are recovered, experts from the early 1990s would need to be consulted in order to decipher them. These experts are continually retiring from the DOE, ensuring that the likelihood of significant additional capability loss increases dramatically with every passing year.

1.2. Determination of the state of knowledge of SFR safety

Between 2009 and 2011, the Advanced Reactor Concepts programme within DOE has funded a series of five gap analysis panels tasked with identifying safety related gaps that still remain in the knowledge needed for making the safety case for licensing an SFR. Both burner and breeder reactors were considered in this expert elicitation, although their respective needs were not delineated in the report's recommendations. Filling these gaps would be essential in order to license a future SFR. It was expected that, because of the relatively mature state of SFR technology, many of the identified gaps would be related to design options that were near maturity at the end of the Integral Fast Reactor programme, i.e. metallic fuel qualification, evolution of the licensing structure (e.g. increased use of probabilistic risk assessment), or because of loss of institutional knowledge (e.g. abandoned computer codes).

Expert panel elicitation was used in order to identify the significant regulatory gaps in each of the five topical areas. Experts were asked to rank each research area on the technical adequacy of existing knowledge in an area of interest and the importance of the research area to the licensing process.

The five topic areas examined for safety related gaps included:

— **Accident sequences and initiators**: How well known are the accidents and associated phenomena that are important in establishing the safety case for licensing an SFR? [1].
— **Sodium technology**: How well can a designer accommodate and model potential energetics associated with sodium fires? [2].
— **Fuels and materials**: How well does the existing experimental database allow for fuel qualification and use of advanced structural materials, and what is the status of fuel performance computer codes? [3].
— **Source term characterization**: How well can we model the source term for an SFR to support emergency planning and other regulatory issues? [4].
— **Codes and methods**: What are the status and capabilities of existing computer codes and models for SFR accident analysis? [5].

1.3. Objective

The report which this paper summarizes is intended both to make research recommendations based on five previously conducted safety related gap analysis reports and to determine cross-cutting needs that exist throughout the SFR related DOE complex [6]. While the eventual funding decisions will be privy to the changing needs and budget priorities of external decision-makers, the identification of cross-cutting gaps, i.e. a coordinated knowledge management and preservation effort, may be the most important information highlighted by this report.

In general, this report assumes that a US SFR design will use metallic fuel, either binary (U-10%Zr) or ternary (U-x%Pu-10%Zr). While some of the gap reports considered other fuel types, such as oxide fuel, for simplicity, these gaps are not included in this cumulative report due to the existing US research direction for SFRs. Additionally, it is assumed that gaps relating to design alternatives, e.g. qualifying extremely high burnup fuel, deploying a loop or pool design, or developing a supercritical CO_2 (S-CO_2) power conversion cycle, will have a lower priority associated with resolving outstanding licensing and safety issues than gaps that cross-cut almost any SFR design. Unlike gaps relating to oxide fuel, optional gaps are not discarded from this report’s analysis.

In an attempt to ensure that the research plan identified in this report addresses the needs of all potential US SFR designers, a wide range of stakeholders will be consulted to provide input on the recommendations of this report. A series of case studies were used to assist the decision maker in consideration of a range of varying funding priorities, i.e. low cost or time sensitive gaps.

A draft of this report was released to the DOE in February 2012 (without external input). The final version of the full report was released in March of 2012. This final version includes all external feedback acquired between the release of the draft and final document.

2. HISTORY OF SFR DEVELOPMENT AND US LICENSING

2.1. History of SFR deployment

Next generation nuclear energy systems currently under consideration aim for significant advances over existing and evolutionary light water reactors (LWRs) in the areas of sustainability, economics, safety, reliability and non-proliferation. Development of these systems is an international effort, involving collaborations under the framework of the Organization for Economic Cooperation and Development's (OECD's) Generation-IV International Forum (GIF) and the IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO).

Recent studies under these programmes highlight the importance of closed fuel cycle systems using fast neutron reactors to meet the sustainability goals through efficient resource utilization. In comparison to LWRs, fast reactors can extract about two orders of magnitude more energy from the same amount of fuel. Moreover, nearly all long lived heavy elements (transuranic waste that remains radioactive for a long time) can be consumed in a fast reactor with a closed fuel cycle, greatly reducing the amount of repository space needed for waste isolation.

Although reconsidered as part of the next generation of nuclear reactors, the fast-spectrum systems, particularly the liquid SFRs, are not new concepts. Since the 1950s, SFR technologies have been pursued and demonstrated worldwide, leading to the construction and operation of several experimental and prototype fast reactors in France (RAPSODIE, Phénix, and Superphénix), Germany (KNK and SNR-300), Japan (JOYO and Monju), the United Kingdom (DFR and PFR), the former USSR (BR-10, BOR-60, BN-350 and BN-600), and the USA (Experimental Breeder Reactor (EBR-I and –II), FERMI, Southwest Experimental Fast Oxide Reactor (SEFOR), and the Fast Flux Testing Facility

(FFTF)). These fast reactors have achieved well over 300 reactor-years of accumulated operation experience.

Although interest in SFRs has declined during the past several decades, the restart of Monju in Japan, completion of the Chinese Experimental Fast Reactor (CEFR), and ongoing construction of the Prototype Fast Breeder Reactor (PFBR) in India and BN-800 in the Russian Federation have demonstrated a renewed interest in SFR development. Recent commercial interests in building SFRs within the USA have been shown by GE (Power Reactor Innovative Small Modular (PRISM)), Toshiba (Super-Safe, Small, Simple (4S)), and TerraPower (TP-1).

2.2. Open items from the previous SFR licensing efforts

The current US SFR licensing experience has come about from the Clinch River Breeder Reactor (CRBR) and the Advanced Liquid Metal Reactor (ALMR) programme interactions with the US Nuclear Regulatory Commission (NRC). In the 1970s and early 1980s, the DOE attempted to license CRBR, but Congress cut funding before the project was complete. While core disruptive accidents (CDAs) were not considered as part of the design basis for CRBR, accidents that could lead to CDAs, including unprotected accidents and large break loss of coolant accidents, received a large amount of regulatory attention, which prolonged the licensing process. The NRC's Atomic Safety and Licensing Board (ASLB) eventually excluded CDAs from the licensing basis, with the NRC staff stating:

> "It is our current position that the probability of core melt and disruptive accidents can and must be reduced to a sufficiently low level to justify their exclusion from the design basis accident spectrum."

The CRBR licensing process resulted in an NRC Safety Evaluation Report in 1983: NUREG-0968 [7].

After CRBR was cancelled in 1983, the DOE embarked on the ALMR programme. This programme emphasized a pool type reactor concept and metal fuel to avoid severe accident related regulatory issues that impeded CRBR's licensing. After an initial design competition between the PRISM (GE) and Sodium Advanced Fast Reactor (SAFR) (Rockwell/Westinghouse) reactor concepts, with both designs submitting a Preliminary Safety Information Document (PSID) to the NRC in 1986, the GE-led PRISM reactor became the focus of the ALMR programme in 1988. Coupling the ALMR fuel cycle with GE's PRISM reactor concept later led to establishment of the Integral Fast Reactor (IFR) programme [8].

The resulting Pre-application Safety Evaluation Report (PSER) highlighted key regulatory issues for PRISM [9, 10]. The major non-design-specific items highlighted by the NRC staff in the PRISM PSER include:

- — Limited performance and reliability data for passive safety feature;
- — Unverified analytical tools used to predict plant response;
- — Limited supporting technology and research;
- — Limited construction and operating experience;
- — Incomplete information on the proposed metallic fuel.

While the IFR programme continued to address these identified issues, some of them were never fully resolved due to the abrupt closure of the IFR programme in 1994. While the PSID was analysed by a relatively small group within the NRC and thus cannot be taken as encompassing all potential regulatory issues, the PSER provides the best indication of potential regulatory concerns for the sodium reactor. Many of these issues were also captured by the five supporting gap analyses. Only the limited construction and operating experience was neglected from our findings because this gap can only be addressed through the act of building more reactors, not sponsoring more research.

3. METHODOLOGY

3.1. Selection of the five topical areas

Since SFR design studies are at an early stage and currently include many system design and fuel options, the safety related gap evaluation must address the range of relevant options needed to fully assess the phenomena that must be considered in the safety evaluation. In defining the major safety related gap topic areas, it is recognized that the topic areas and phenomena are driven by the potential accident sequences (the first topic area listed below). The accident sequences are in turn significantly affected by the SFR design. However, some aspects of SFR safety, such as sodium phenomena, are less design specific. A preliminary list of gap topic areas was identified based on general knowledge of SFR technology and requirements for evaluating the SFR safety case. The topic areas (corresponding to panels) for the SFR safety related gap evaluation project are:

— **Accident initiators/sequences**: The accident scenario panel addressed the broadest scope of safety relevant phenomena. This panel utilized available information and expert opinion to define classes of events with safety significance and the systems or subsystems that are affected; identified the phenomena that are active in those events; assessed the importance of these phenomena against safety criteria; and assessed the state of knowledge for analysing the safety significance of these phenomena. The broad spectrum of phenomena addressed in this panel resulted in overlap with the other phenomenological panels. The scope of this panel extended to secondary systems and balance of plant interactions with accident events where appropriate.
— **Sodium phenomena**: Sodium coolants add an additional dimension of safety relevant phenomena that must be considered in the overall evaluation of SFR safety. The probability and location of a sodium fire is design dependent, but the phenomena associated with the range of sodium fires that could result from a leak in the primary or secondary system can be assessed at an early stage. There has been considerable research on sodium fires and sodium phenomena, such as sodium concrete interactions, in the USA and in other countries as part of previous fast reactor development programmes. Pool fire phenomena that were considered by this panel include radiation heat fluxes between the pool surface and environment, aerosol generation, convection at the surface, development of the oxide crust and the sodium flow (spreading) issue. Spray fire phenomena include modelling of the plume and spray dynamics (thermofluid dynamics), spray characteristics, including droplet size and velocity distributions, chemical combustion kinetics, and agglomeration phenomena. Sodium interactions with concrete may also result in hydrogen production and aerosol generation.
— **Fuels and materials**: Understanding the characteristics of fuels and key materials under accident conditions is essential to reactor safety analysis. Advanced fuel characterization under both normal and accident conditions is needed to assess fission product release/retention, fuel–coolant interactions, fuel–clad interactions, fuel swelling, and fuel motion mechanisms under accident conditions. Both metal and oxide fuel types are being considered for SFRs and the implications of both types must be evaluated. The anticipated high burnups and long service lifetimes also pose issues for key non-fuel materials, particularly cladding materials. The very high burnup and the resulting high fast neutron fluences require that clad properties (embrittlement, swelling, etc.) be understood and included in accident analysis. The fuels and materials panel assessed fuel and material phenomenology important to safety and identified the information needed to support the overall safety and licensing approach. Fuel types and

materials of construction, and the associated conditions of safety relevant service, would be design and scenario dependent, and therefore this panel was convened after the accident initiator panel.

— **Source term**: The source term of primary interest is the release of radionuclides to the site and beyond the site boundary. To assess the defence-in-depth of a plant design, the regulatory process will also define a source released to the containment and evaluate the leakage of radionuclides to the environment. The concentrations of radionuclides suspended in the containment atmosphere as functions of time are of crucial importance. With the exception of noble gas releases and some small fraction of radioactive iodine release, the radionuclides are suspended in the reactor containment as aerosol particles. Aerosol sources to the containment arise directly from fuel in the case of fuel handling accidents. Otherwise, important aerosol releases to the containment come from the sodium coolant. The least intense type of radionuclide release from sodium comes from quiescent sodium pools retained in the reactor coolant system. Contamination of the sodium coolant with radionuclides is most important when fuel rods have ruptured and the fuel is exposed to the coolant.

— **Computer codes and models**: Based on the range of scenarios and phenomenology identified from the SFR safety evaluation performed in the other safety related gap topics, the codes and models panel addressed the analytical capabilities and data required to adequately assess the safety implications of SFR scenarios and phenomena. The scope of this panel included the assessment of thermohydraulics, heat transfer, and structural and neutronics modelling capabilities, as well as the evaluation of the validation basis. Of particular interest is the evaluation of accident analysis tools that are generally unique to nuclear reactor safety. In addition, this panel addresses the potential for modern advanced modelling and simulation techniques to improve nuclear safety analysis approaches using higher fidelity, integrated multi-process tools. This activity was closely coupled to the accident initiators/sequence panel scope.

3.2. Selection of expert panels

While each of the five gap analyses used slightly different criteria to select their expert panels, the following three guidelines were generally followed:

— The panel should be chaired by an authority in the topical area of interest.

— The panel should include at least one expert in every topical area analysed.

— The panel should be representative of the DOE complex and, if possible, the international community.

The panel size varied for each topic ranging from 5 to 12 panellists.

3.3. Gap identification and ranking process

The individual panel evaluations identified safety relevant features and components that are involved in the range of accident sequences relevant to that panel, and then assessed the phenomena active in those scenarios. The panels assessed the importance of those phenomena to the safety case for an SFR and the knowledge level currently available to address these issues for licensing. Gaps or areas of inadequate understanding were identified to define safety related R&D needs. Figure 1 shows a high level description of how the gap analyses were conducted.

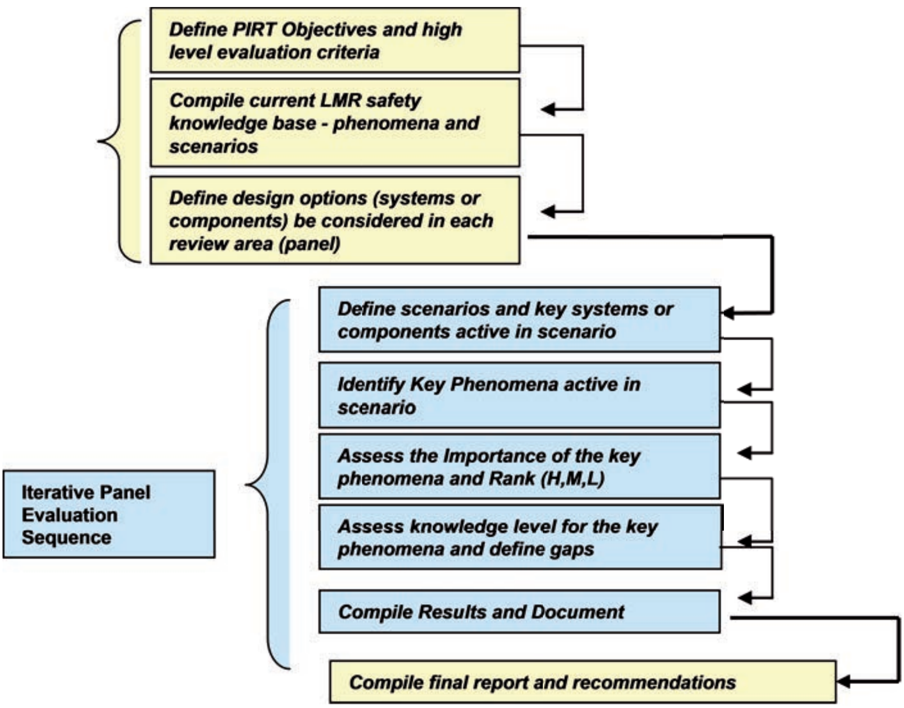


FIG. 1. Sequence of gap analysis activities and panel process.

The importance of the issues identified by each panel was ranked qualitatively by the panel members as of either: high (H), medium (M), or low (L) importance. The general descriptions of these importance ranking levels are:

— High (H) — phenomena are of first order (fundamental) importance based on evaluation criteria.
— Medium (M) — phenomena are of secondary (contributing) importance based on evaluation criteria.
— Low (L) — phenomena are not important for the scenario and evaluation criteria being considered.

Evaluating the state of knowledge of a phenomenon generally involves the assessment of both the modelling capabilities and the database to validate the model(s). The panels discussed each phenomenon extensively during the evaluation, with the general criteria for ranking the state of knowledge defined as:

High (H)

— A physics based or correlation based model is available that adequately represents the phenomenon over the parameter space of interest.
— A database exists adequate to validate relevant models or to make an assessment.

Medium (M)

— A candidate model or correlation is available that addresses most of the phenomenon over a considerable portion of the parameter space.
— Data are available but are not necessarily complete or of high fidelity, allowing only moderately reliable assessments.

Low (L)

— No model exists, or model applicability is uncertain or speculative.
— No database exists; assessments cannot be made reliably.

The gap analysis knowledge results are also provided in the summary table, which includes comments for each ranking. In that same section of the report, we provide details of the rationale or justification for the panel knowledge ranking in our discussion.

4. OBSERVATIONS AND RECOMMENDATIONS

This report was intended to make research recommendations based on five previously conducted safety related gap analysis reports. While the eventual adoption path for funding recommendations will be subject to the changing needs and budget, the identification of cross-cutting gaps, that is, a coordinated knowledge management and preservation effort, may be the most important information highlighted by this report.

4.1. Review of the project

This paper began with a history of SFR involvement by the DOE and around the world. Historical licensing efforts with the CRBR, SAFR and PRISM were summarized. While omitted from this paper, the full report discusses how the increasing adoption of probabilistic elements to the licensing framework may change the level of regulatory scrutiny applied to a future licensing effort. This shifting emphasis is reflected in the proposed SFR regulatory standard, ANSI/ANS 54.1. These regulatory shifts must be understood when starting new research and development initiatives to improve the licensability of an SFR.

Four of the five underlying topic areas considered in this gap analysis are the same as those identified to improve the licensability of the next generation nuclear power programme. Because of the specific safety concern related to handling sodium, a fifth topic area relates to sodium phenomenology. These five gap analyses were led by recognized experts in their respective fields and participated by panels of additional experts to identify regulatory hurdles facing the topical area. These panels also attempted to rank the current state of knowledge and regulatory significance of each gap. In an attempt to ensure that the results were design neutral, any optional design feature considered was evaluated on the basis that that design feature would be included in the licensing case. It was left to the writers of this report to flag these design optional gaps and for the eventual decision maker to determine the relative importance of funding safety and licensing related issues in these research areas. It is the authors' opinion that these gaps, while they ultimately may be required to ensure economic feasibility of the design, should be a lower priority than gaps that will affect all SFR designs. Design optional gaps that improve the economic efficiency of a design should not be completely ignored, but instead should only be procured through cost sharing partnerships with interested commercial partners.

4.2. Observations from prioritization scenarios

As a result of the gap evaluation process discussed above, six budget priority scenarios were developed to help decision makers determine which gaps get funded in the near term. The following observations were seen to be cross-cutting or high priority issues for each of the five gap reports.

Accident sequences and initiators

— Knowledge preservation and management: While independent efforts are currently under way throughout the SFR community to secure and analyse historical databases, these efforts are only sporadically coordinated and often underfunded. A DOE-NE led effort to ensure that the historical database is not lost or destroyed will be instrumental in supporting any future licensing effort.

— Owing to economic considerations, any gap requiring high burnup metallic fuel is not likely to be filled in the foreseeable future. Without domestic fast neutron irradiation facilities, any new irradiations would need to be conducted in one of the few remaining international fast reactors. Owing to host restrictions, these experiments would most likely be limited to pin scale tests. In order to remove regulatory hurdles associated with the identified high burnup gaps, full assembly and/or full core experiments will be required.

Sodium technology

— Some US facilities are currently under-utilized and are capable of addressing many high-priority gaps. It may be cost effective for the DOE to fund domestic or to host internationally funded experiments in these under-utilized facilities before they are closed due to lack of funding.

— There is currently no maintained capability within the DOE for modelling of sodium fire phenomenology. This gap is related to the containment response gap discussed in source term characterization below.

Fuels and materials

— Consistent with the comment on knowledge preservation and management in accident sequence and initiators, the user-base and expertise for the fuel performance and licensing code LIFE-Metal computer code has almost disappeared. Currently, one steward is relied upon to keep the code accessible to future users. This situation is untenable; thus, sufficient funding should be provided in the near term to train the next generation of users not only how to operate the code, but how to explain and defend the code to a licensee or regulator.

— Owing to economic considerations, any gap requiring future irradiation of metallic fuel to high burnup is not likely to be filled within the next 10–15 years.

Source term characterization

— There is currently no maintained capability for modelling of the containment response and estimation of the corresponding source term. The extension of capabilities under a severe accident code such as MELCOR for SFRs will be important to support any licensing effort. The passive safety of current SFR designs may not avoid the need for a containment code.

— While source term related research requiring irradiated fuel will likely need to be postponed until SFR funding levels increase, research requiring only radionuclide tracers, e.g. radionuclide release from fuel debris into a quiescent sodium pool and radionuclide behaviour in containment, can be conducted using existing facilities at Idaho National Laboratory and Oak Ridge National Laboratory. Because the accidents in Fukushima will likely require that the SFR source term form part of an advanced reactor licence application, it is logical to begin research in this area in the near to mid-term in order to avoid severe delays if a SFR licence is submitted.

Codes and methods

— See MELCOR comment in source term characterization.

— Owing to Fukushima, experimental data concerning the response of SFR core materials and structures, systems and components to earthquakes and other external events need to be collected and used to improve current computational models.

— The state-of-the-art concerning transitions from full power to natural circulation will need to be improved to defend passive safety as a layer within defence-in-depth.

— SAS4A needs to be modernized and improved (in terms of modelling accuracy, functionality and usability), and adapted to modern software engineering practices.

Common issues

— The current process for handling of documenation with the access control designation applied technology was determined to be at best complicated and at times counterproductive. The current process makes removing applied technology designations on documents which no longer need to be protected extremely difficult. Additionally, the NRC is not set up to handle applied technology documents within its current knowledge management system. This makes any applied technology document unusable to a licensee. The DOE-NE needs to develop a new process for streamlining the applied technology process if an SFR licence is submitted.

4.3. Final remarks

Most gaps associated with SFRs are related to either the loss of historical data and capabilities or to new technologies designed to make SFRs economically competitive. It is assumed that much of the historical experimental database from the IFR safety and licensing will be licensed with metallic fuel clad with stainless steel, in either a pool or a loop configuration, with a Rankine power conversion cycle, and with a burnup within the range demonstrated by metallic fuel. Variations of this theme have been constructed multiple times in both the USA and internationally, if both metallic oxide fuel forms are considered. A more aggressive design, with different cladding options, higher burnups, possible use of transuranic fuel elements or targets and advanced power conversion cycles, will likely require a new irradiation testing facility be built.

After completion of this report in March of 2012, the DOE has initatied programmes to both modernize SAS4a and incorporate Contain-LMR into the MELCOR code system.

ACKNOWLEDGEMENTS

The authors would like to express their appreciation to the Advanced Reactor Concepts (ARC) Program within DOE-NE for funding this multi-year effort. This report was finished under the DOE-ARC work packages # AR-12SN040205, AR-12IN040205, AR-12OR040205, and AR-12BN040205.

The authors would also like to thank the chairs and members of the supporting gap analysis reports for their continued assistance and recommendations for resolving the gaps their panels identified. During this report, Thomas Wei (Argonne National Laboratory (ANL)), Richard Vilim (ANL), Richard Wood (Oak Ridge National Laboratory (ORNL)), and David Holcomb

(ORNL) were tasked with identifying gaps in instrumentation and control for sodium reactor technology. Their assistance in this regard was invaluable. The expertise of Chris Grandy (ANL), Ken Natesan (ANL), and Abdellatif Yacout (ANL), was vital in helping compile the ranking tables in this report. Gratitude is also extended to Jesse Philips (SNL), John Sackett (INL), and Art Wright (ANL), for their assistance in editing the report.

The authors would like to thank TerraPower, Westinghouse, and AREVA, for their feedback regarding the results of the five gaps analysis reports. Their industrial insight was greatly appreciated.

Sandia National Laboratories is a multi-programme laboratory managed and operated by Sandia Corporation, a wholly owned subsidiary of Lockheed Martin Corporation, for the US Department of Energy's National Nuclear Security Administration under contract DE-AC04-94AL85000.

REFERENCES

[1] SACKETT, J., et al., Advanced Sodium Fast Reactor Accident Sequences Technology Gap Analysis, Rep. FCR&D-REAC-2010-000126, DOE Fuel Cycle Research and Development Program, Washington, DC (2010).

[2] CORRADINI, M., et al., Advanced Burner Reactor Sodium Technology Gap Analysis, Rep. FCR&D-REAC-2010-000034, DOE Fuel Cycle Research and Development Program, Washington, DC (2010).

[3] WALTERS, L., et al., Sodium Fast Reactor Fuels and Materials: Research Needs, Rep. SAND2011-6546, Sandia Natl Lab., NM (2011).

[4] POWERS, D.A., et al., Advanced Sodium Fast Reactor Accident Source Terms: Research Needs, Rep. SAND2010-5506, Sandia Natl Lab., NM (2010).

[5] SCHMIDT, R., et al., Sodium Fast Reactor Gaps Analysis of Computer Codes and Models for Accident Analysis and Reactor Safety, Rep. SAND2011-4145, Sandia Natl Lab., NM (2010).

[6] DENMAN, M., et al., Sodium Fast Reactor Safety and Licensing Research Plan Vol. I, Rep. SAND2012- 4260, Sandia Natl Lab., NM (2012).

[7] UNITED STATES NUCLEAR REGULATORY COMMISSION OFFICE OF NUCLEAR REACTOR REGULATION, Safety Evaluation Report Related to the Construction of the Clinch River Breeder Reactor Plant, NUREG-0968, Vols 1 and 2, Washington, DC (1983).

[8] IVANS, W., The Current Nuclear Regulatory Environment: A PRISM Perspective. M.S. Thesis,. University of Florida, Gainesville, FL (2006).

[9] KING, T.L., UNITED STATES NUCLEAR REGULATORY COMMISSION OFFICE OF NUCLEAR REACTOR REGULATION, Preapplication Safety Evaluation Report for the Sodium Advanced Fast Reactor (SAFR) Liquid-Metal Reactor, NUREG-1369, Washington, DC, (1991).

[10] UNITED STATES NUCLEAR REGULATORY COMMISSION OFFICE OF NUCLEAR REACTOR REGULATION, Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Reactor, NUREG-1368, Washington, DC (1994).

IAEA SAFETY STANDARDS FOR FAST NEUTRON REACTORS AND HIGH TEMPERATURE GAS COOLED REACTORS

N. TRICOT
International Atomic Energy Agency,
Vienna

P. CHELLAPANDI
Indira Gandhi Centre for Atomic Research,
Kalpakkam, India

F. REITSMA
Calvera Consultants,
Strubensvallei, South Africa

K. TUČEK
European Commission, Joint Research Centre,
Institute for Energy and Transport,
Petten, Netherlands

Abstract

Present considerations of innovative, next generation reactor concepts with improved safety and sustainability characteristics have motivated the IAEA to launch a series of consultancy meetings to interpret applicability and make a proposal for a possible revision of its Safety Standards for these advanced reactor designs. This paper summarizes results of the first step taken to critically review the applicability of IAEA Safety Standards Series No. NS-R-1, Design of Nuclear Power Plants, and IAEA Safety Standards Series No. GSR Part 4, Safety Assessment for Facilities and Activities. Efforts have been made to generalize the safety requirements with due consideration given to the specific safety features that are associated with fast reactors and high temperature gas cooled reactors. Insights from the Fukushima Dai-ichi events have also been considered to incorporate lessons learned.

1. INTRODUCTION

The designs of many existing nuclear power plants, as well as the designs of new plants expected to be operational in the near to mid-term future have been enhanced to include additional measures to prevent and mitigate the consequences of complex accident sequences involving multiple failures and of severe accidents. The design of new nuclear power plants now explicitly includes consideration of severe accident scenarios and strategies for their management and mitigation (Lever 4 of Defence-in-Depth). This, together with the more recent need to incorporate lessons learned from the Fukushima Dai-ichi events, has motivated a review of Safety Requirements publications, including IAEA Safety Standards Series No. NS-R-1, Design of Nuclear Power Plants [1], now superseded by IAEA Safety Standards Series No. SSR-2/1 [2], and IAEA Safety Standards Series No. GSR Part 4, Safety Assessment for Facilities and Activities [3].

Despite the previous efforts to improve and update the above requirements in technology neutral terms, these focus mainly on light water reactor technologies. This is also acknowledged in the stated objective [1]:

"It is expected that this publication will be used primarily for land based stationary nuclear power plants with water cooled reactors designed for electricity generation or for other heat production applications (such as district heating or desalination). It is recognized that in the case of other reactor types, including innovative developments in future systems, some of the requirements may not be applicable, or may need some judgement in their interpretation."

The goal of the ongoing review and update effort of safety requirements is therefore to:

— Develop IAEA Safety Standards in more technology neutral terms;
— Include, whenever possible, other types of reactor.

To this end, the IAEA implemented in 2010 and 2011 a series of consultancy meetings to interpret the applicability of the aforementioned Safety Requirements for fast reactor (FR) and high temperature gas cooled reactor (HTGR) technologies. The review of the Safety Requirements for Design of Nuclear Power Plants was based on an earlier version of the Standard, NS-R-1, and a draft version of SSR-2/1.

The detailed objectives of the consultancy meetings were:

— To review and discuss potential updates to the safety requirements for safety assessment and for design;
— To make proposals for necessary changes or additions to include new types of reactors;
— To suggest a roadmap for the implementation of these changes into the standards;
— To interpret the safety requirements in view of the needs identified for FRs and HTGRs.

During the consultancy meetings, these two requirements were critically reviewed with reference to their application to FRs and HTGRs, starting a process for the preparation of detailed interpretations and comments for each requirement as well as for providing suggestions for possible adaptations. As such, an initial attempt has been made to generalize the safety requirements and their interpretation. In this process, due consideration has also been given to specific design and safety features that are associated with FRs and HTGRs.

In the case of FRs, the aspects addressed included those related to the coolant (the chemical reactivity with water and air and associated chemical toxicity effects, opacity, freezing, boiling, etc.) and to the core, such as high power density, control rod worth, core compactions, and material issues such as corrosion and erosion. In addition, the special safety requirements for the fuel handling processes and systems, more specifically the decontamination aspects, were also considered.

In the case of the HTGRs, the passive cooling function that does not rely on the presence of the helium coolant, the containment/confinement choice and the possible release of helium to the environment in the case of a large break (very fast depressurization), the sharing of auxiliary systems important to safety in a multi-unit plant and the safety aspects associated with graphite (prevention of air ingress and corrosion of the pressure boundary) formed the basis for the proposal of the main modifications and interpretations.

A short summary of the work is provided in this paper with the focus on the FR related comments and interpretations compared to those proposed for HTGRs. Parts of the summary have already been published earlier with the focus on HTGRs [4].

In Section 2, background information on the IAEA Safety Standards and related activities is provided. In Section 3, a representative selection of the requirements, where interpretation for FRs as well as HTGRs is specifically required, is highlighted and discussed. This is followed by conclusions and references.

2. BACKGROUND

The IAEA has developed a system of safety documentation that not only reflects an international consensus of best practices, but also constitutes the basis for achieving a high level of safety for protecting people and the environment from the harmful effects of ionizing radiation. On the highest level are the fundamental safety principles supported by the safety requirements [5] (see Fig. 1).

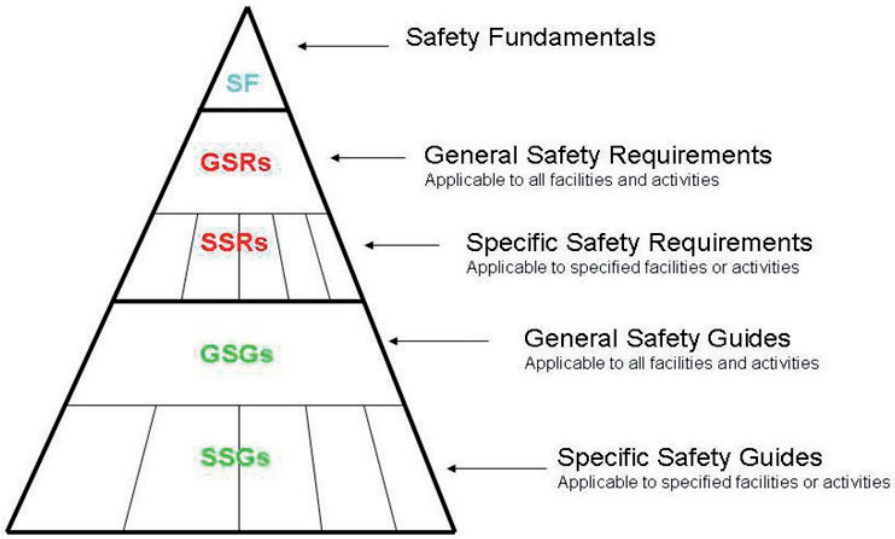


FIG. 1. IAEA Safety Standards categories.

The Safety Fundamentals (SF-1), the General Safety Requirements (GSR), and the General Safety Guides (GSG) are applicable to all facilities and activities. These are complemented by Specific Safety Requirements (SSR) and Specific Safety Guides (SSG), which are applicable to specified facilities and activities. The transition to this new structure (adopted in 2008) is currently under way.

As an example, in 2000 the IAEA published NS-R-1 Safety Requirement (considered in this review) which was recently updated in order:

— To improve its content, taking into consideration other newly published Safety Standards;
— To consider the feedback received from Member States and from the Design Safety Review Services;
— To keep the standard updated with the evolution of the technology and nuclear safety and to assure a high level of safety.

The new Requirement was published in 2012 as SSR-2/1 [2].

3. REQUIREMENT EXAMPLES

A summary of some of the main requirements that were found in need of the interpretation or possible modification is given below.

3.1. Codes and standards

The requirement stipulates that items important to safety will be designed in accordance with the relevant national and international codes and standards. Therefore, the establishment of codes and standards (mostly not available for FRs and HTGRs) might be needed. The development of such standards may require research and development and a step-by-step approach (prototypes versus proven design). As an example for liquid metal cooled reactors (LMRs), generic codes for the design for aerosol deposition and minimization, gas entrainment, cold trap, in-service inspection and repair, material corrosion and erosion in lead or lead–bismuth eutectic coolants, etc., are not available.

The acceptance of these codes and standards is made under the supervision of national regulatory authorities. It is therefore beneficial if some advice and guidance based on experiences in Member States can be included in the establishment of the codes, standards and guidelines. As an example, in the FR programme in India, the codes and standards are identified, evaluated and reviewed by a committee of experts, which submit their opinion on applicability, adequacy and sufficiency to the safety authority. In some cases, the committee itself can develop some specific rules, codes and standards.

3.2. Multi-unit and co-located facilities

Aspects related to multi-unit plants as well as to co-located facilities have also been considered. These are common for LWRs, FRs and HTGRs, and require special attention for small and modular designs and installations with process heat applications, including hydrogen production.

Many requirements on the control room (and supplementary control room) are included in NS-R-1. The sharing of a control room in the case of small modular reactors is not directly addressed, but is very relevant in discussions today. There are designs for reactor units having a common control room since they are connected to a single power turbine in a single plant. This is the case with the HTR-PM design where two reactors will be connected to a single generator [6]. Multiple units are also envisaged in the future. The small LWR based units of Nuscale [7] also propose a multi-module configuration where each module operates independent of the others, but is managed from a single control room.

Modular, passively safe FR designs are also considered, such as the sodium cooled GE Hitachi Nuclear Energy's Power Reactor Inherently Safe Module (PRISM) [8] and Toshiba Corporation's Super-Safe, Small and Simple (4S) reactor [9], liquid bismuth eutectic cooled SVBR-75/100 [10] and Gen4 Energy Module [11], as well as the lead cooled Secure, Transportable, Autonomous Reactor (STAR) family of reactors: Small STAR (SSTAR) and STAR-LM (LM = liquid metal) [12]. In the case of SFRs, one of the concerns of sodium fire is dispersion of sodium aerosol from health safety considerations. Hence, it is important to respect the aerosol limits in the building specified by national regulations. The sources of aerosol generation could be from sodium fire in the buildings where sodium circuits are located. Within the reactor containment building, the source could be the release of sodium consequent to core disruptive accident. Under such a severe accident situation, the control room designs need to take into account the sodium aerosol dispersion and acceptable limits for the operators. As for the lead–bismuth cooled reactors, specific risks might exist regarding the production of toxic ^{210}Po.

FRs [13] and HTGRs [14] can also be applied to process heat applications, especially due to the higher temperatures available. Nuclear power plants coupled to and located close to an industrial site brings some additional licensing and safety related considerations [15]. In the case of a process heat plant for hydrogen production, the external hazard due to hydrogen explosion has to be taken into account. These may include measures on the distance between the plants and the barriers between the nuclear reactor and the chemical process. These could be added in the future as a general requirement for these applications.

3.3. Corrosion

In certain instances it was also proposed that additional requirements or sub-clauses should be added. One such proposal was under the requirement of the reactor coolant system with a specific focus on corrosion. The proposed new sub-clause reads: "The design of the reactor coolant system shall be such that the possibility and/or consequences related to the ingress of corrosive material or substances leading to corrosive material shall be minimized."

The ingress of water (from the steam generator tube rupture) is an example of the generation of corrosive products that can enter the cooling system (and the reactor) or of the potential mechanism for the loss of a corrosion control in the coolant. This is a requirement particularly important for lead and lead–bismuth cooled fast reactors (LFRs) since at high temperature, flowing lead and lead–bismuth are corrosive to structural materials and can, via physical chemical processes, induce or accelerate material failure under static loading via reduction in ductility and strength or under time dependent loading involving fatigue and creep. This requires operating LFRs at low temperature range and maintaining a controlled concentration of dissolved oxygen in the coolant, which has to be high enough to support the formation of a protective layer of magnetite (Fe_3O_4) on the surfaces of structures and, at the same time, low enough to prevent the formation of large amounts of PbO precipitation. Excessive amounts of PbO and other impurities in the lead coolant might lead to circuit fouling and slagging and possible reduction of the coolant flow cross-sections, potentially causing coolant flow blockages. In SFRs, purity of the coolant also needs to be monitored and controlled to limit the corrosion, even though the requirements on corrosion control are less stringent than for lead cooled reactors.

Design measures to detect, prevent and mitigate the effects of water ingress are implemented in LMR designs [16].

A similar concern is also present for HTGRs. The pressure boundary (and thus the reactor cooling system boundary) must also limit the possibility of large amounts of air ingress and consequent corrosion of graphite in the HTGR. In particular the possibility of two breaks occurring (top and bottom) that can lead to a chimney effect must be minimized.

3.4. Opacity

Unlike LWRs and gas cooled reactors (HTGRs and gas cooled fast reactors (GCFRs)), the coolant of LMRs is optically opaque, which makes maintenance, testing, repair, replacement, inspection and monitoring difficult.

In LMRs, in-service inspections are carried out by ultrasonic devices, for which the technology has been developed in the context of SFR programmes. In view of the possibility of strongly exothermic chemical reactions (sodium with water and air in SFRs), access for maintenance of components operating in liquid metal is very much limited and calls for advanced techniques in case it is demanded. Special care has to be taken in order to limit the exposure to chemical toxic effects due to deposited sodium on the structural wall, liquid metal vapours and aerosols. R&D in these domains is performed intensively.

Fuel handling operations are also difficult and need to be performed remotely. This has been shown to be an important operational aspect, as demonstrated by the experience gathered from fuel handling incidents, which occurred in the Fast Breeder Test Reactor (FBTR) in India as well as in JOYO and MONJU SFRs in Japan.

For that reason, the upper ends of the subassemblies in some LFR designs are placed above the free level of lead in the cover gas [16]. As such, refuelling operations can be done without the need for in-vessel machines.

3.5. Core geometry changes

Since the FR core is not in the most reactive configuration, any small geometrical variations (e.g. external excitation due to an earthquake) could introduce significant reactivity variations. The fuel subassemblies and the supporting structures therefore need to be designed so that in operational states and in accident conditions other than severe accidents, a geometry that allows for adequate cooling is maintained and that the insertion of control rods is not impeded. Consequently, core configuration should not be changed significantly to cause any unacceptable reactivity variations under the above conditions. In particular, core restraint design (concept and location on the subassembly wrapper) should be done in such a manner as to introduce minimum reactivity variation.

Core compaction in the case of pebble bed reactors, as well as ease of insertion of control rods into the channels, are the issues to be addressed in graphite based HTGRs.

3.6. Freezing of coolant

Unlike in LWRs and gas cooled reactors, the coolant freezing might impact the fulfilment of fundamental safety function with respect to reactivity control (impairing control rod functions) and heat removal (forming coolant blockages). Consequently, the coolant needs to be kept molten during normal operating conditions, including the shutdown situations, handling operations and emergency situations, such as a turbine trip. The issue is relevant for sodium (T_m = 98°C), lead–bismuth eutectic (T_m = 125°C), but of particular importance for lead (T_m = 327°C) cooled systems.

During the shutdown states (in particular for handling operations), the coolant temperature must be continuously monitored and it also must therefore be heated, when the decay heat is not sufficient enough.

For SFRs, frictional heat due to circulation of the coolant in the primary circuit is enough to avoid coolant freezing (sodium temperature in the range of 150°C). For LFRs, during the shutdown states and maintenance outages decay heat given off by the radioactive isotopes of the core is enough to keep the lead above its melting point. However, at beginning-of-life conditions or during long outage periods, when decay heat is not enough to keep lead molten, an auxiliary heating system is included in the LFR designs, in order to ensure the minimum temperature of the lead by transmitting heat from the secondary system [16].

For LMRs, the liquid metal coolant pipelines are provided with heaters to maintain the temperature of coolant above melting point all the time, even under shutdown conditions and further, the purity of coolant (control of oxygen in liquid metal) is maintained, so that there is no concern of flow blockage in the coolant path.

3.7. Boiling

Sodium boiling may result in voids which could introduce positive void coefficient in SFR. Hence, special provisions should be made and appropriate safety criteria should be evolved to ensure the reactor safety.

3.8. Operating experience and feedback

In terms of reliability of items important to safety and in the absence of adequate long term commercial operating experience (~400 reactor-years for SFRs, specifically in China, France, Germany, India, Japan, the Russian Federation (the former USSR), the United Kingdom and the United States of America, ~80 reactor-years for lead–bismuth cooled reactors in the former USSR, and ~50 reactor-years for HTGRs in China, Germany, Japan, the UK and the USA), key plant parameters such as temperatures and design life have to be chosen carefully. They have to be enhanced in a step by step manner with the accumulated experience and reliability demonstrated.

There is no operating experience and feedback available on lead cooled and GCFRs.

3.9. Fuel design differences

In the design of the core, there are different types of concepts such as fuel form (pellets, rods, plate, pebbles), fuel geometrical arrangement (bundle, can, sheath), clad type (standalone, collapsible), configuration (horizontal, vertical) and operating conditions (different levels of burnup and temperatures) depending on the type of reactor (LWR, SFR, LFR, GCR, MSR). In Fig. 2, the typical LWR, two different HTGR fuel designs, and the SFR fuel are shown. Towards evolving generic safety design terms and criteria, there is a need to homogenize the safety design requirements which is applicable for all types of fuel. For a specific reactor type, such terms and associated requirements would be interpreted as applicable. In this respect, terms such as fuel pins, fuel elements and subassemblies are used in NS-R-1 and need to be either generalized or interpreted for specific reactor types. In a pebble type HTGR, a fuel element can be the fuel sphere or pebble; and for a prismatic design the fuel compact may perhaps be equated to the fuel pin. The specific reference to the fuel cladding (as typical in the LWR fuel design) is another example. In the event of a loss of coolant accident, the limiting parameters of the fuel cladding may not be exceeded so as to minimize fuel damage and limit

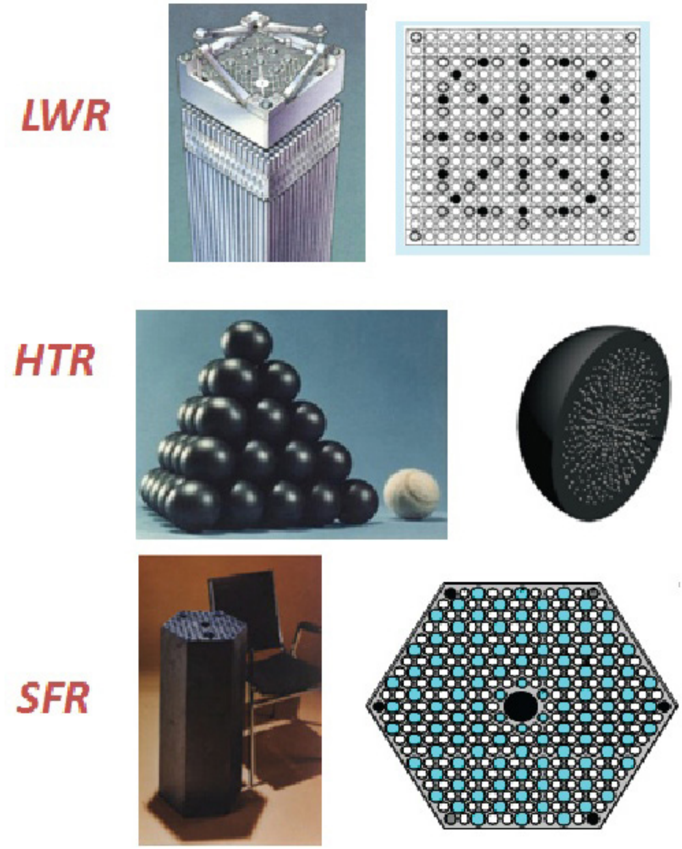


FIG. 2. Different fuel designs [17].

the release of fission products from the fuel. In the multiple barrier and defence-in-depth examples, the fuel cladding is also specifically mentioned.

One way to generalize the requirements may be to equate these ‘LWR terms’ to the typical ‘HTR terms’ as suggested above (‘pin’ ≡ ‘compact’). The glossary or definitions could be expanded, or specific interpretations of the existing terms can be made for HTGR fuel designs. In many cases, this may introduce more confusion in the broader sense when other safety requirements are considered — an example follows.

It is typical to refer to multiple pins or subassembly failures when core degradation or core damage is defined. This is also typically linked to the cladding melting temperature and emergency core cooling.

This will lead to misinterpretation when a coated particle fuel in a modular HTGR is considered since the fuel failure mechanism and consequences of failures are different. The failure of the fuel pin and its cladding is closely related to LWRs or FRs, whereas the failure of a fuel compact or fuel sphere would not be defined in the same context for HTGRs. In this case, a failure of a large fraction of the coatings of the coated particles could perhaps be seen as a similar event to fuel pin/cladding failure in LWRs and FRs. This example illustrates that a simple generalization of terms and definitions may not be adequate, but rather that interpretations of the different requirements may have to be written and agreed upon.

In other aspects such as the handling and storage of fuel, the existing requirements may also be too specific. Whereas care should be taken not to drop LWR and FR fuel subassemblies or an HTGR prismatic fuel block, the pebble fuel is dropped into the reactor on fuel load and also dropped in the spent fuel casks or spent fuel storage tanks when unloaded. A generalization to describe the requirements as to prevent mechanical (and other) impacts beyond it design limits, may be more appropriate and applicable to all fuel designs.

Pebble bed reactors have an additional requirement that damaged fuel spheres (based on a specific criterion, e.g. on dimension) should be removed by the fuel handling system which may not be applicable to other designs.

The reasons for the stated requirement to permit inspection of the fuel in the fuel handling and storage system must also be understood. In a pebble bed design with large number (possibly millions) of fuel spheres and where fuel is either stored in larger spent fuel tanks on-site or in transport flasks, it may be very difficult to achieve. Cover atmosphere sampling may, for example, be done for the fuel in the container (but not individually).

3.10. Considerations on Fukushima Dai-ichi events

In the review, preliminary insights with respect to the Fukushima Dai-ichi events, in particular concerning risks and consequences related to the beyond design basis earthquake, beyond design basis flooding, and the total loss of the power and the ultimate heat sink were also considered.

The consequences of a large earthquake might be core deformations and compactions leading to possible loss of reactivity control, loss of integrity of the main and/or safety vessels or loss of the pressure in primary system of HTGRs and GCFRs. Specifically, in view of the large specific mass of lead, the response of lead and lead–bismuth cooled reactors to earthquakes needs to be carefully considered.

Post-Fukushima analysis has motivated investigation of the severe accident scenarios critically, postulating various load combinations. In LMRs, the failure of primary coolant pressure boundary (main vessel in the case of pool type designs) will not lead to a complete loss of decay heat removal capability due to presence of a safety vessel surrounding to the main vessel. In SFRs, another primary coolant pressure boundary is formed by primary pipes (‘LIPOSO’). Rupture of one of the pipes is a design basis event (Category-IV) and it can be demonstrated that the design safety limits are not exceeded under this event. In loop type reactors, the entire primary circuit has a double envelope and in the case of GCFRs, a similar concept is adopted by the provision of the guard vessel. Even in the case of failure of both main vessel and guard vessel, the complete loss of coolant could be prevented by including required structural features, such as sodium resistant concrete with external cooling provisions, in the reactor vault. Thus, the loss of integrity of the primary coolant pressure boundary is less stringent for FRs.

For SFRs, however, specific consideration must be given to the consequences of possible multiple failures leading to sodium leakages and sodium fires or large sodium–air–water interactions, which might lead to damage of rooms or components.

The ingress of water/steam to the core needs to be practically excluded for SFRs in view of the abovementioned exothermic reaction of sodium with water, the consequences of which might be difficult to manage. LFRs, GCFRs and HTRs are more tolerant to water ingress, but major water ingress also needs to be practically excluded as it might lead to positive reactivity insertions together with possible water chemistry effects.

For LMRs, including SFRs, inherent characteristics related to coolants having a high degree of thermal inertia and a capability for natural convection provide increased robustness to the total loss of electric power supply and/or the heat sinks. As such, decay heat removal function can be achieved either fully or at least partially passively. On the other hand, GCFRs have very low thermal inertia, which in the case of the depressurization of the primary circuit requires that either forced circulation is maintained within the primary circuit or a minimum adequate pressure is kept to ensure that sufficient natural circulation is maintained.

For modular HTRs, the decay heat removal is performed by passive means through the natural phenomena of conduction, convection and radiation and thus does not rely on any active system or even the presence of the helium coolant. The building and surrounding earth can serve as the ultimate heat sink or this function can be fulfilled by the reactor cavity cooling system, which limits the temperatures of the concrete structures and which can be designed to operate in a passive mode. In modular HTRs which prescribe very low power densities and large surface areas of a metal vessel to radiate heat, so-called core meltdown is practically eliminated.

4. CONCLUSIONS

During the evaluation it was found that most of the design requirements for the safety of nuclear plants are directly applicable to the evaluated FR and HTGR designs without the need for any modifications. The specific requirements are also relatively easily adaptable and expandable to include FRs and HTGRs. It was however concluded that many of the requirements need specific interpretation or guidelines for the design specific aspects.

As such, a technology neutral requirement will require designers to focus on the safety performance instead of on existing, sometimes concept specific solutions (for LWRs).

Much more effort, consultations as well as involvement of other stakeholders are required to develop a technology neutral reactor design safety requirement supported by safety guides for each type of reactor.

ACKNOWLEDGEMENTS

The work reported is based on the discussions and thoughts in consultancy meetings sponsored by the IAEA. This is an ongoing IAEA task that is not fully completed and whose final results may be published as an IAEA TECDOC.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design, Safety Standards Series No. NS-R-1, IAEA, Vienna (2000).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design, Safety Standards Series No. SSR-2/1, IAEA, Vienna (2012).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for Facilities and Activities, Safety Standards Series No. GSR Part 4, IAEA, Vienna (2009).

[4] REITSMA, F., CHELLAPANDI, P., TUČEK , K., TRICOT, N. , “Safety Requirements for Design and Safety Assessment of HTGR and Fast Reactors”, HTR2012-8-002, paper presented at 6th Int. Topical Mtg on High Temperature Reactor Technology (HTR2012), Tokyo, Japan, 2012.

[5] TRICOT, N., “Input for Development of Technology Neutral IAEA Safety Standards”, paper presented at at 22nd Mtg of the IAEA Technical Working Group on Gas Cooled Reactors (TWG-GCR-22), IAEA, Vienna (2011).

[6] ZHANG, Z., WU, Z., WANG, D., XU, Y., SUN, Y., LI, F., DONG, Y., Current status and technical description of Chinese 2 × 250 MWth HTR-PM demonstration plant, J. Nucl. Eng. Des. **239** (2009) 1212.

[7] HAUGH, B., MOHAMED, A., "Comparison of a Nuscale SMR Conceptual Core Design Using CASMO5/SIMULATE5 and MCNP5", Proc. Int. Conf. Advances in Reactor Physics (PHYSOR) Knoxville, TN, 2012, American Nuclear Society, LaGrange Park, IL (2012) CD-ROM.

[8] GE HITACHI NUCLEAR ENERGY (GEH), PRISM Sodium-cooled Reactor (2014),http://gehitachiprism.com/

[9] Toshiba Corporation's Super-Safe, Small and Simple Reactor (4S), Advanced Reactors Information System, IAEA, Vienna, Austria, http://aris.iaea.org/ARIS/reactors.cgi?requested_doc =report&doc_id=76&type_of_output=html

[10] ZRODNIKOV, A.V., CHITAYKIN, V.I., TOSHINSKY, G.I., GRIGORIEV, O.G., DRAGUNOV, U.G., STEPANOV, V.S., KLIMOV, N.N., KOPYTOV, I.I., KRUSHELNITSKY, V.N., GRUDAKOV, A.A., "SVBR-75/100 — Lead-bismuth cooled small power modular fast reactor for multi-purpose usage", IAEA-CN-108/36, paper presented at. Int. Conf. on Innovative Technologies for Nuclear Fuel Cycles and Nuclear Power, Vienna, Austria, 2003, IAEA, Vienna (2003)

[11] GEN4 ENERGY INC., http://www.gen4energy.com/

[12] WADE, D.C., "The STAR concept: A hierarchical hub-spoke nuclear architecture based on long refuelling interval battery reactors and regional fuel cycle centres", Innovative Small and Medium Sized Reactors: Design Features, Safety Approaches and R&D Trends, IAEA-TECDOC-1451, IAEA, Vienna (2005) 171.

[13] YILDIZ, B., KAZIMI, M.S., Efficiency of hydrogen production systems using alternative nuclear energy technologies, Int. J. Hydrogen Energy, **31** (2006) 77.

[14] DELANNAY, N., et al., "EUROPAIRS prospects for coupling of HTR with industrial processes: viability assessment", paper presented at 5th Int. Topical Mtg on High Temperature Reactor Technology (HTR2010), Prague, Czech Republic, 2010.

[15] VERFONDERN, K., NISHIHARA, T., Valuation of the Safety Concept of the Combined Nuclear/Chemical Complex for Hydrogen Production with HTTR, Rep. Jül-4135, Forschungszentrum Jülich, Germany (2004).

[16] ALEMBERTI, A. et al., these Proceedings.

[17] Workshop on Advanced Reactor Concepts: Very High Temperature Reactors (VHTR); Presented on 15 April 2012 at the PHYSOR2012 meeting, Knoxville, TN, USA; http://physor2012.org/workshops.html, 2012.

FAST REACTOR MATERIALS: ACHIEVEMENTS AND NEW CHALLENGES

Chairpersons

P. DUBUISSON
France

C. FAZIO
Germany

DEVELOPMENT OF STRUCTURAL MATERIALS FOR THE JSFR: OVERVIEW AND CURRENT STATUS

T. ASAYAMA, T. KAITO
Japan Atomic Energy Agency,
Ibaraki, Japan

Abstract

This paper summarizes the ongoing efforts regarding new core and structural materials that will be applied to the Japanese Sodium Cooled Fast Reactor (JSFR). For core materials, oxide dispersion strengthened ferritic steel and 11Cr ferritic-martensitic steel will be applied. For structural materials, 316FR stainless steel and modified 9Cr-1Mo steel will be applied. The current status of alloy design, acquisition of the data necessary to establish the material strength standard, fabrication techniques to meet the requirements of the design of the JSFR and codification of the material strength standards regarding the new materials are overviewed. Further described is the path forward to the application of the materials in the JSFR.

1. INTRODUCTION: FACT PROJECT AND MATERIAL DEVELOPMENT

The Japanese Sodium Cooled Fast Reactor (JSFR) is a demonstration reactor that follows the prototype reactor Monju. The JSFR adopts a number of innovative designs in order to improve safety, reliability and economic competitiveness [1]. New core materials for fuel pins and duct tubes, and new structural materials for vessels and piping are necessary to meet the requirements and are being extensively developed [2].

Fuel pin cladding and duct tubes of the JSFR fuels will be irradiated to peak neutron dose of 250 dpa and exposed to peak burnup of 250 GW·d/t. In order to improve thermal efficiency of the plants, maximum outlet coolant temperature at the reactor vessel is determined to be 823 K, and the corresponding maximum (hot spot) temperature of the cladding tube is 973 K and that of the duct tube is 843 K. For achievement of high burnup, core materials are required to have good dimensional stability under a high dose neutron irradiation environment. Thus, both tubes should be made of ferritic steels having excellent swelling resistance. Ferritic steels for core application are roughly divided into two types: oxide dispersion strengthened (ODS) ferritic steels and conventional ferritic/martensitic (F/M) steels. It is necessary for the cladding tubes to be ODS ferritic steels, because F/M steels will rapidly lose their strength over 923 K. On the other hand, F/M steels can be applied to the duct tubes. ODS ferritic steel and 11Cr F/M steel (PNC-FMS) have been selected as the most prospective candidate materials for cladding and duct tubes, respectively. Manufacturing technology developments will be extended from laboratory scale to commercial scale. The in-reactor and out-of-reactor mechanical tests of these steels will be continued in order to prove their in-reactor performance and codify their material strength standard. As for the out-of-reactor tests, material irradiation tests will be continued in the Joyo experimental fast reactor. Additionally, fuel pin irradiation test and bundle irradiation tests will be conducted in the Joyo experimental fast reactor.

The reactor vessel and internal structures of the JSFR will be made from 316FR, which is austenitic stainless steel with superior creep-fatigue resistance and will allow for a compact design. 316FR was developed in Japan and based on SUS316 of the Japanese Industrial Standard, which is equivalent to Type 316SS [3]. This material has already been applied to the intermediate heat exchanger of the Joyo experimental fast reactor. For the coolant system, including primary and secondary piping, intermediate heat exchangers and steam generators, Mod.9Cr-1Mo steel, which is equivalent to ASME Grade 91 steel, will be used. The high thermal conductivity and low thermal expansion of this material significantly contribute to shortening the piping system and simplifying the design of components. The maximum temperature of the coolant for normal operation of the primary system is 823 K and the design life is 60 years. Pathways to apply the two materials to the JSFR are categorized into several technical areas. The first one is acquisition of long term data that form the basis for the 60-year design and the development of evaluation methods of material performance of which a typical example is creep-fatigue evaluation methodologies. The second point is the development of manufacturing technology to produce components unique to the JSFR, which include a large diameter forged ring for the reactor vessel (316FR), very long and thin-walled heat exchanger tubes for intermediate heat exchangers and steam generators (Mod.9Cr-1Mo steel) and very thick forged plate for the tubesheets of steam generators. The technical development will be codified in the Japan Society of Mechanical Engineers (JSME) Code for design and construction of fast reactors. The 2012 edition of the JSME Code newly incorporates 316FR and Mod.9Cr-1Mo steel with time dependent allowable stresses for basic product forms up to 300 000 hours [4, 5]. The 2016 edition will extend them to 500 000 hours and will include product forms unique to the JSFR. This paper describes some of the main features of the above pathways.

2. CORE MATERIALS FOR FUEL ASSEMBLIES

2.1. ODS steel

2.1.1. Material development

Nanometre size complex oxide dispersoids in matrix provide excellent dispersion strengthening for ODS ferritic steels. In the JAEA, two types of ODS ferritic steel have been developed for the cladding tubes: ODS tempered martensite and ODS recrystallized ferrite. ODS tempered martensitic steel aims at higher radiation resistance with a basic chemical composition of Fe-0.13C-9Cr-2W-0.2Ti-0.35Y_2O_3 in wt% (9Cr-ODS tempered martensite). In addition, from the viewpoint of the improvement of corrosion resistance, e.g. fuel compatibility and corrosion property in nitric acid solutions and so on, development of the ODS tempered martensitic steel having the Cr concentration higher than 9wt% is now being carried out to increase the flexibility of development. ODS recrystallized ferritic steel aims at higher corrosion resistance with a basic chemical composition of Fe-0.03C-12Cr-2W-0.3Ti-0.23Y_2O_3 in wt% (12Cr-ODS recrystallized ferrite), although irradiation induced hardening and embrittlement at low temperatures are a concern for 12Cr-ODS. We decided from the viewpoints of formability and irradiation performance, etc., that ODS tempered martensite should be the primary candidate [6], and that manufacturing technology development for mass production will be performed primarily for 9Cr-ODS tempered martensitic steel.

2.1.2. Fabrication and verification

The manufacturing process of the ODS ferritic steel cladding tubes is described in Fig. 1. The manufacturing process of ODS ferritic steel cladding tubes can be done via a pre-mix process, which refers to the fabrication process using the elemental powder (Fe, Cr, C, W, Ti) and Y_2O_3 powder as raw material powders; or in an alternative, via a pre-alloy process, which refers to the manufacturing process using the Ar gas atomized alloy powder and Y_2O_3 powder, which is further classified in partial pre-alloy process and the full pre-alloy process. Raw material powders are mechanically alloyed in an Ar gas atmosphere by using a 10 kg attriter. The mechanically alloyed powders were then filled into mild steel cans, degassed and hot extruded at 1423 K. The extruded bars were machined and drilled to mother tubes. The ODS ferritic steel cladding tubes 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 transformation from

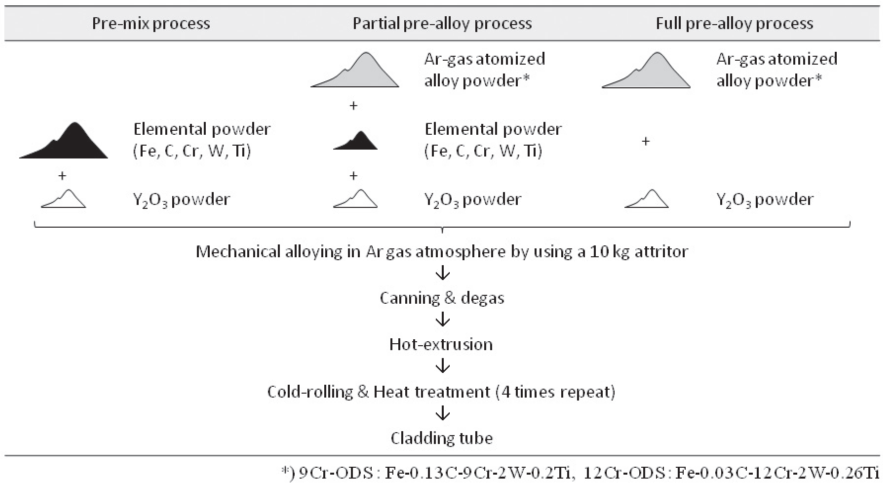


FIG. 1. Manufacturing process of ODS steel cladding tubes.

α to γ in 9Cr-ODS tempered martensite and recrystallization in 12Cr-ODS ferrite to reduce strength anisotropy by grain morphology control [7, 8]. Mechanical alloying and hot consolidation processes dominate the dispersoid morphology and resultant mechanical properties.

In order to evaluate the mechanical properties of manufactured ODS ferritic steel cladding tubes, ring tensile tests and creep rupture tests with internal pressure were carried out. Manufactured ODS ferritic steel claddings showed improved tensile strength over the entire temperature region and the uniform elongation was adequately maintained. Figure 2 shows the internal creep rupture strength in comparison to PNC-FMS and the 20% cold worked modified 316 stainless steel (PNC316). The internal creep rupture strength attained 120 MPa for 10 000 hours at 973 K which is a target required for JSFR fuel design. This strength level is much better than that of PNC-FMS and even superior to PNC316 under the stress condition to be usually used as fuel pins. It is evaluated for strength levels of cladding, not by creep tests but by temperature-transient-to-burst in the case of transient events.

To investigate the in-reactor performance of ODS ferritic steels for use as fuel cladding tubes, and establish a material strength standard for fuel pin mechanical design, hundreds of specimens and dozens of fuel pins have been irradiated in Joyo and in BOR-60. As a result of material irradiation tests executed in Joyo, the irradiation data concerning the tensile properties of ODS steel cladding tubes were obtained at temperatures between 673 and 1108 K up to a neutron dose of 33 dpa. Figure 3 shows the experimental results. There was no

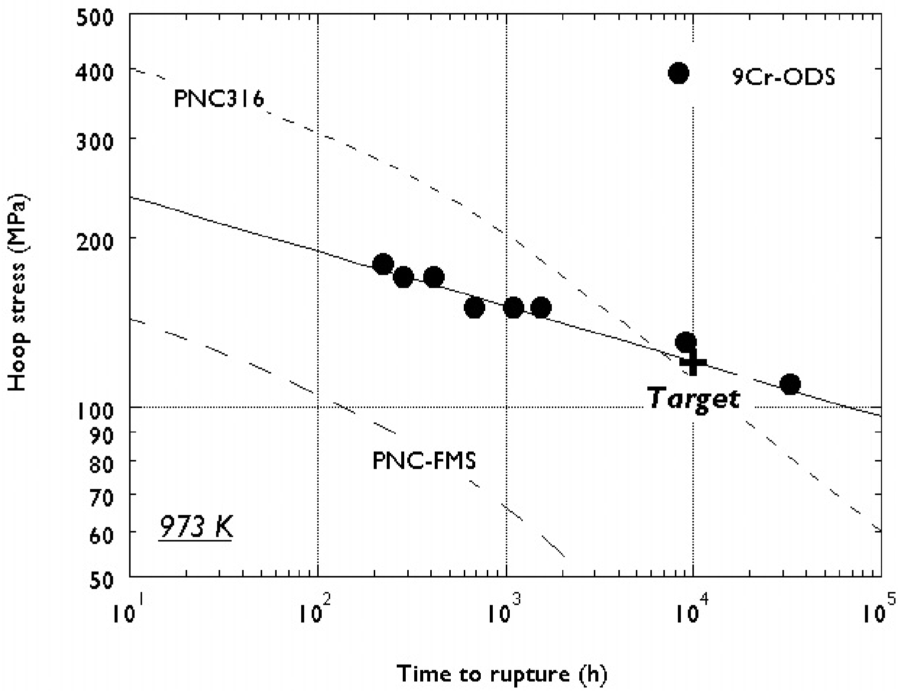


FIG. 2. Creep rupture strength of manufactured 9Cr-ODS tempered martensitic steel cladding tubes.

significant degradation in tensile strength even after neutron irradiation [9]. The in-pile creep rupture tests using pressurized tube specimens were also carried out at Joyo. As a result, the irradiation effects on creep rupture strength were negligible up to a neutron dose of 3.1 dpa and for 614 hours [10].

The BOR-60 irradiation test had been completed under a collaborative programme between the JAEA and the Research Institute of Atomic Reactors. The experimental fuel assemblies containing 9Cr-ODS tempered martensitic and 12Cr-ODS recrystallized ferritic steel cladding fuel pins were irradiated up to peak burnup of 112 GW·d/t and peak neutron dose of 51 dpa. Superior properties of the ODS claddings concerning fuel compatibility, dimensional stability under irradiation and so on were confirmed as shown in Fig. 4. On the other hand, peculiar irradiation behaviours such as microstructure instability and fuel pin rupture occurred in 9Cr-ODS tempered martensitic steel cladding tube. The cause of these peculiar irradiation behaviours was concluded to be the combined effects of two factors: the matrix Cr heterogeneity in the 9Cr-ODS tempered martensitic steel fuel cladding tube and the irradiation temperature exceeding the designed limit. The 9Cr-ODS tempered martensitic steel cladding tubes used for the

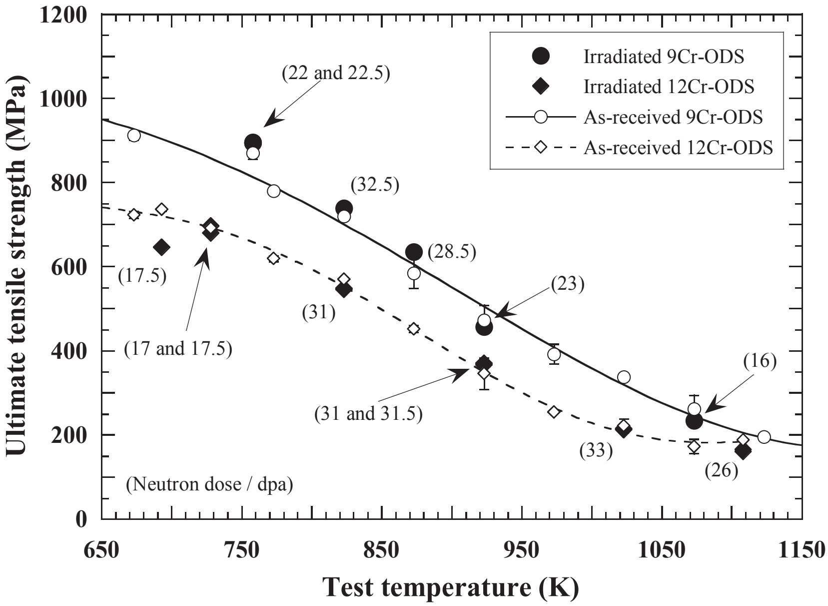
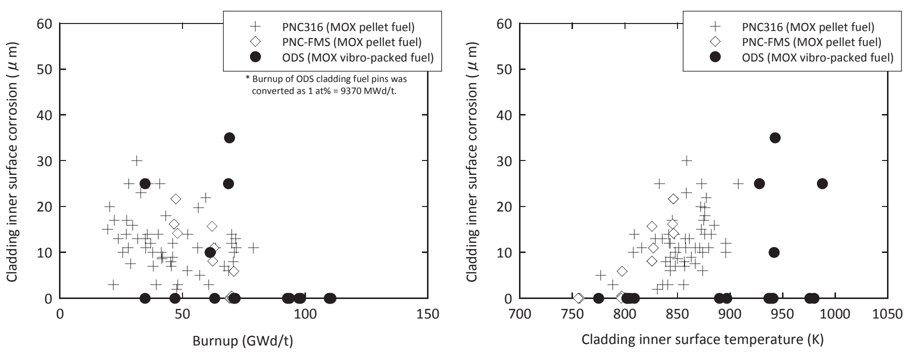


FIG. 3. Ultimate tensile strength of irradiated ODS steel cladding tubes.



(a) Burnup dependence (b) Temperature dependence

FIG. 4. Fuel compatibility of the irradiated ODS claddings compared with the FCCI of PNC316 and PNC-FMS claddings.

BOR-60 irradiation test were manufactured by the pre-mix process. The handling of elemental powder in the production process increases the difficulty in assuring composition homogeneity of the cladding tube. To improve homogeneity of 9Cr-ODS tempered martensitic steel, the JAEA has started developing the production technology based on the full pre-alloy process.

2.2. 11Cr F/M steel

Since the maximum temperature of the duct tubes is expected to be about 843 K, PNC-FMS was selected as the primary candidate duct material. We have 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 wt%. PNC-FMS specimens were irradiated in Joyo and in the Fast Flux Test Facility to prove its validity and now PNC-FMS has been applied to the duct tube of reflectors in Joyo MK-III cores.

SUS316 is applied to entrance nozzle and handling head of the fuel subassembly in the JSFR. The development of the PNC-FMS duct tube with SUS316 short joints at the both ends has been carried out to manufacture the fuel assembly. The dissimilar fusion welding process using electron beam welding was selected as the primary candidate technology. Currently, mechanical tests of PNC-FMS/SUS316 specimens joined by electron beam welding have been carried out to evaluate the integrity of the welded parts.

3. STRUCTURAL MATERIALS FOR REACTOR VESSEL AND PIPING

Austenitic stainless steel 316FR will be used for the reactor vessel and internals to realize a compact design with superior resistance to elevated temperature use. Mod.9Cr-1Mo steel, which is basically equivalent to ASME Grade 91 steel, will be adopted for the primary and secondary coolant systems including intermediate heat exchanges and steam generators. The high heat conductivity and low thermal expansion ratio of the material enable a shortened piping system and compact design of intermediate heat exchangers and steam generators. Both materials will be subjected to 823 K at maximum, for 60 years, which is the design life of the JSFR. The present status of material development, long term data acquisition and technology development for the fabrication regarding these materials, along with the codification efforts within the Japan Society of Mechanical Engineers are summarized in this chapter.

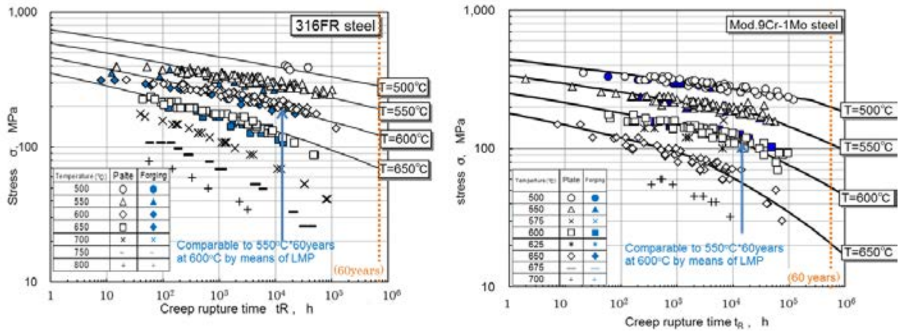
3.1. Data acquisition and evaluation of long term properties

316FR was developed in Japan for liquid metal cooled fast reactors. The base material is SUS316 of the Japanese Industrial Standard which is equivalent to Type 316SS. The chemical composition was optimized to improve creep properties by reducing carbon content and adding nitrogen. A unique point with 316FR is that phosphorus is also added to improve creep properties. Table 1 shows the chemical composition of 316FR.

The characteristics of the materials have to be captured from various aspects including not only mechanical properties but also environmental effects and long term ageing. From this viewpoint, an extensive database has been established for both materials. The data consist of basic mechanical properties such as fatigue data, long term creep data as well as in-sodium fatigue and creep-fatigue tests, tests with aged materials, post-irradiation tests and tests with some complex loading histories, such as a combination of prior fatigue and subsequent creep tests.

The most important failure mode to be prevented in the design of fast reactors is creep-fatigue. Therefore, long term creep data, which form the basis for creep-fatigue evaluation is very important. Creep data were obtained to cover temperature ranges up to 1073 K for 316FR and 973 K for Mod.9Cr-1Mo. The maximum rupture time is over 100 000 hours, as shown in Fig. 5. Time to rupture was formulated as a function of stress and temperature using the Larson-Miller parameter. For Mod.9Cr-1Mo steel, in order to improve the accuracy of regression in a long term region, the region splitting analysis method [11] has been adopted; the rupture curve was determined for the short term and long term regions, respectively, using a second order polynomial function. The intercept of the two curves was determined as half of yield strength at temperature. This method significantly improved the accuracy of regression in the long term region [12, 13].

Creep-fatigue tests were also performed and available data were collected focusing on conditions closer to practical applications which are low strain range and long hold time. The lowest strain range, the maximum hold time and the duration to failure were 0.3%, 100 hours and 91 181 hours, respectively. Based on the fatigue data and creep data thus obtained, the applicability of the creep-fatigue damage evaluation procedure in the JSME Code, which was developed for conventional materials such as SUS304, was verified for the new materials. Fatigue damage was calculated by the Robinson Rule and creep damage was calculated by the time fraction approach. A creep-fatigue damage envelope with an interception of (0.3, 0.3) was employed for both materials. Figure 6 indicates the result that as time to creep-fatigue failure increases, conservatism, defined as the ratio of predicted creep-fatigue life to measured



(a) 316FR (b) Mod.9Cr-1Mo steel

FIG. 5. Long term creep rupture tests.

creep-fatigue life, remains constant for 316FR and tends to increase in the case of Mod.9Cr-1Mo steel [14, 15].

Another point of significant importance for the JSFR design is fabrication and evaluation of welded joints. Tests with a similar scheme to those mentioned above are performed concurrently for welded joints of the two materials, including dissimilar joints of the two materials. In the case of Mod.9Cr-1Mo steel, close attention is paid to Type IV cracking and creep-fatigue evaluation methodologies that take account of it are being developed.

3.2. Fabrication of products for the JSFR

As described in Section 1, there are some specifications unique to the JSFR with regards to material product forms. This section shows the technology development for the manufacturing and the forgings for the tubesheets of the steam generators. In order to increase the capacity of the steam generators, the size of the tubesheets will be approximately 3000 mm in diameter and 800 mm in thickness. This is much larger than the size conventionally fabricated, particularly as regards thickness. Therefore, fabrication technology was verified by producing samples that simulate the actual production process. The points to be verified and ensured were negligible segregation, negligible anisotropy of mechanical properties and stability of microstructure (evolution of martensite). A test product with the same height and half the diameter of the actual forging was made. In order to suppress segregation of coarse niobium, the electroslag remelting method was employed. Almost the same forging ratio as the actual forging was applied. Normalizing temperature and hold time were 1323 K and 22 hours, and temperature and hold time were 1033 K and 17 hours. The hold times were longer than those of thin products.

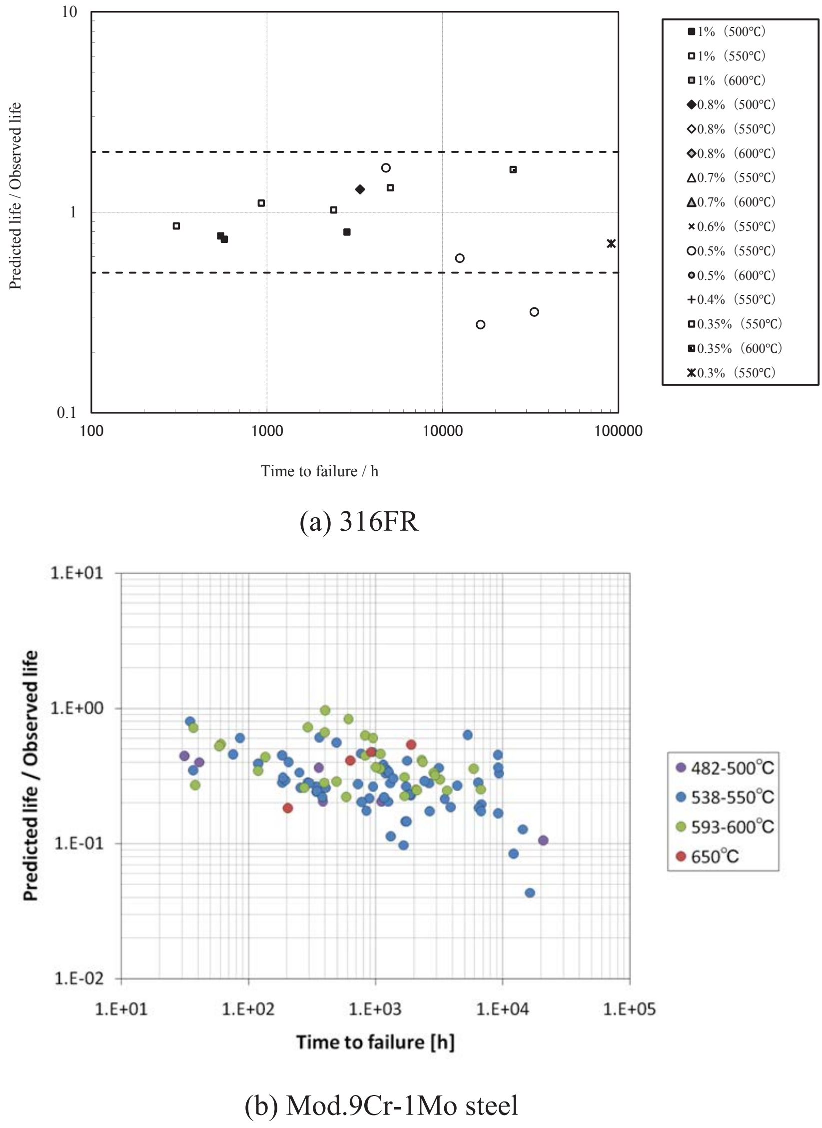


FIG. 6. Long term creep-fatigue life prediction.

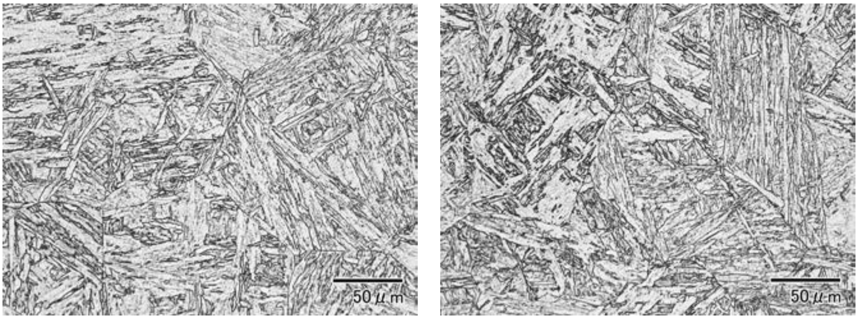
The chemical composition of the test product in Table 2 shows that segregation was negligible. Tensile properties were almost uniform in the product irrespective of location and it was revealed that anisotropy of tensile properties is small. Microstructure was martensitic both at the centre of the forging and at the position of half-thickness, as shown in Fig. 7. Therefore, the fabrication

TABLE 1. CHEMICAL COMPOSITION OF 316FR (wt%)

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

TABLE 2. CHEMICAL COMPOSITION OF MOD. 9CR-1MO STEEL (wt%)

	C	Si	Mn	P	S	Ni	Cr	Mo	V	Nb	Al	N
	0.08–0.12	0.20–0.50	0.30–0.60	0.020 max	0.010 max	0.40 max	8.00–9.50	0.85–1.05	0.18–0.25	0.06–0.10	0.04 max	0.030–0.070
Test product	0.10	0.24	0.48	0.008	0.001	0.28	8.65	0.92	0.19	0.07	0.008	0.053



(a) At the center of the forging (b) At the position of half-thickness

FIG. 7. Microstructure of thick-forging.

technology has been verified in terms of microstructure and short term mechanical properties. Long term tests such as creep and creep-fatigue have just started.

3.3. Codification in the JSME

For the design and construction of the JSFR, the JSME Code for design and construction of fast reactors will be used. The latest 2012 edition newly incorporates 316FR and Mod.9Cr-1Mo steels. A set of allowable stresses were prepared in the same manner as conventional materials with time dependent ones up to 300 000 hours. Equations such as creep rupture equation, creep strain equation, equations for low cycle fatigue life and so on were also determined. An evaluation method for neutron irradiation effects was incorporated for 316FR. Sodium environmental effects are incorporated for both 316FR and Mod.9Cr-1Mo steels. In the course of the adoption of the new materials, a number of structural tests were performed and applicability of the elevated temperature design methods to the new materials was verified [15].

The 2016 edition of the JSME Code will incorporate time dependent allowable stresses up to 500 000 hours to enable a 60-year design. Product forms with specifications unique to the JSFR, such as the very thick forgings of Mod.9Cr-1Mo steels, will also be codified.

4. FUTURE PERSPECTIVE

4.1. Core materials

Manufacturing technology for mass production should be developed for both steels. For commercialization of ODS ferritic steel cladding tube, there are two critical issues in the production technology: the technological transition to commercial scale and the improvement of tube homogeneity. Thus, JAEA will start the production technology development for scale-ups in mechanical alloying and hot extrusion processes along with improving the production process based on the full pre-alloy process. For PNC-FMS, the manufacturing technology of the dissimilar welding duct tube will be established. The irradiation performance of both steels will be evaluated through material, fuel pin and bundle subassembly irradiation tests in Joyo. These data will be used for upgrading of the material strength standards for the JSFR fuel.

4.2. Structural materials

In order to ensure structural integrity for 60 years, a practical scheme which consists of 3 methodologies is being developed. The first one is acquisition of very long term creep data, including rupture time data covering 200 000 hours and more. Some of the tests have already started and could continue after the operation of the JSFR. The second point is quality assurance of materials from the viewpoint of long term change of mechanical properties associated with degradation of microstructures. From this viewpoint, extrapolation methods for creep and creep-fatigue based on the fracture energy approach [16] is being investigated. The third point would be to monitor material degradation in the actual plant with either non-destructive or destructive methods. Electromagnetic methods are being developed for this purpose.

5. SUMMARY

Regarding the core materials, the high burnup capability of the JSFR fuels depends significantly on irradiation performance of their component materials. The JAEA has been developing ODS ferritic steels and PNC-FMS as the most prospective materials for fuel claddings and duct tubes, respectively. We will continue to develop manufacturing technology for mass production to supply these steels for future SFR fuels. Mechanical properties of the products were derived by out-of-reactor and in-reactor tests including material irradiation tests in the Joyo experimental fast reactor.

For structural materials, 316FR for the reactor vessel and Mod.9Cr-1Mo steel for the primary and secondary coolant systems will be newly used for the JSFR. Long term material data acquisition has been continued and creep data over 100 000 hours at 823 K, for example, have been successfully generated, and these efforts will be further continued to minimize uncertainties associated with extrapolation. Fabrication technologies will be developed for products special in terms of their size and configuration, which are required to achieve the JSFR's superior performance. The 2012 edition of the JSME Code for design and construction of fast reactors has been published. The 2016 edition will extend the time dependent allowable stresses up to 500 000 hours.

REFERENCES

[1] SATO, K., et al., A preliminary assessment of the adoption of innovative technologies in the Fast Reactor Cycle Technology Development (FaCT) project in Japan, Energy Procedia **7** (2011) 140–145.

[2] ASAYAMA, T., NAKAGIRI, T., WAKAI, T., "Development of structural materials for Japanese demonstration fast breeder reactor", Proc. Workshop on Structural Materials for Innovative Nuclear Systems (SMINS), Karlsruhe, Germany, 2007, OECD Publishing, Paris (2008).

[3] NAKAZAWA, T., et al., Advanced type stainless steel 316FR for fast breeder reactor structures, J. Mater. Process. Technol. **143–144** (2003) 905–909.

[4] JAPAN SOCIETY OF MECHANICAL ENGINEERS, Design and Construction Code Part II Fast Breeder Reactors, JSME S NC-1 (2012).

[5] ASAYAMA, T., et al., "Development of 2012 Edition of JSME Code for Design and Construction of Fast Reactors (1) Overview", Proc. American Society of Mechanical Engineers (ASME) Pressure Vessels and Piping Conference, Paris, 2013 (2013) PVP2013-98061.

[6] UKAI, S., KAITO, T., OHTSUKA, S., NARITA, T., SAKASEGAWA, S., "Development of optimized martensitic 9Cr-ODS steel cladding", Transactions of the 2006 American Nuclear Society Annual Meeting 2006, Reno, NV (2006).

[7] UKAI, S., KAITO, T., OHTSUKA, S., NARITA, T., FUJIWARA, M. and KOBAYASHI, T., ISIJ Int. **43** (2003) 2038–2045.

[8] OHTSUKA, S., UKAI, S., FUJIWARA, M., KAITO, T. and NARITA, T., J. Nucl. Mater. **329–333** (2004) 372–376.

[9] YANO, Y., OGAWA, R., YAMASHITA, S., OHTSUKA, S., KAITO, T., AKASAKA, N., INOUE, M., YOSHITAKE, T., TANAKA, K., J. Nucl. Mater. **419** (2011) 305–309.

[10] KAITO, T., OHTSUKA, S., INOUE, M., ASAYAMA, T., UWABA, T., MIZUTA, S., UKAI, S., FURUKAWA, T., ITO, C., KAGOTA, E., KITAMURA, R., AOYAMA, T., INOUE, T., J. Nucl. Mater. **386–388** (2009) 294–298.

[11] KIMURA, K., KUSHIMA, H., ABE, F., J. Key Eng. Mater. **171–174** (2000) 483–490.

[12] ONIZAWA, T., et al., "Development of 2012 Edition of JSME code for design and construction of Fast Reactors (2) Development of the material strength standard of 316FR stainless steel", Proc. American Society of Mechanical Engineers (ASME) 2013 Pressure Vessels and Piping Conference, Paris, 2013 (2013) PVP2013-97608.

[13] ONIZAWA, T., et al., "Development of 2012 Edition of JSME Code for Design and Construction of Fast Reactors (3) Development of the material strength standard of modified 9Cr-1Mo steel" Proc. American Society of Mechanical Engineers (ASME) Pressure Vessels and Piping Conference, Paris, 2013 (2013) PVP2013.

[14] NAGAE, Y., et al., "Development of 2012 Edition of JSME Code for Design and Construction of Fast Reactors (5) Creep-fatigue evaluation method for 316FR stainless steel", Proc. American Society of Mechanical Engineers (ASME) Pressure Vessels and Piping Conference, Paris, 2013 (2013) PVP2013-97806.

[15] ANDO, M., et al., "Development of 2012 Edition of JSME Code for Design and Construction of Fast Reactors (6) Design margin assessment for the new materials to the Rules", Proc. American Society of Mechanical Engineers (ASME) Pressure Vessels and Piping Conference, Paris, 2013 (2013) PVP2013-97803.

[16] NAGAE, Y., Evaluation of creep-fatigue life based on fracture energy for modified 9Cr-1Mo steel, J. Mater. Sci. Eng. A **560** (2013) 752–758.

DEVELOPMENT OF STRUCTURAL CORE COMPONENTS FOR BREEDER REACTORS

N. SAIBABA
Nuclear Fuel Complex,
Department of Atomic Energy,
Hyderabad, India

Abstract

For India's three stage nuclear power programme, fast breeder reactors (FBRs) will play a crucial role in terms of better utilization of nuclear resources and will have the potential for minor actinide burning which leads to the reduction of both quantity and toxicity of radioactive waste requiring ultimate disposal. The Nuclear Fuel Complex (NFC) in Hyderabad is responsible for the manufacture of all the subassemblies and control elements for the 500 MW(e) Prototype Fast Breeder Reactor (PFBR) being built at Kalpakkam and is also actively involved in the development of materials for future FBRs. The NFC has successfully developed a variety of reactor core components for FBRs. Since performance of fuel clad determines the realization of objectives such as burnup, breeding ratio and linear power rating, there has been continual development in material selection. The NFC has developed the manufacturing process technology for D9 clad tubes and supplied the first full core of the PFBR. A detailed programme has been taken up to establish the manufacturing technology for advanced clad materials such as oxide dispersion strengthened steels. Development of double clad tubes with an outer cover made of 9Cr-1Mo and an inner liner of zirconium finds application as a clad tube with metallic fuels (U-Pu-Zr). Metallic fuel is preferred owing to realization of higher breeding ratio and burnup compared to oxide, carbide and other ceramic fuels. Detailed characterization studies have been carried out after establishing structure property correlation. For double clad tube, seam analysis was carried out to establish proper bonding at the interface of zirconium and 9Cr-1Mo. Hex–Hex cold rolling was adopted for the manufacture of finished hexcan wrapper to ensure the minimum variation of cold work across the cross-section, particularly at the corners. All the subassemblies required for the first breeder power reactor were successfully manufactured by NFC. The welding and machining parameters for various components and subassemblies were optimized. The paper highlights the characterization studies and successful development of components and fuel subassemblies meeting stringent nuclear requirements.

1. INTRODUCTION

Nuclear energy is the most sustainable energy source, not only for India but also for the economic growth of the entire world. India has a very ambitious nuclear power programme for the fast growth of PHWRs and PWRs. Relying

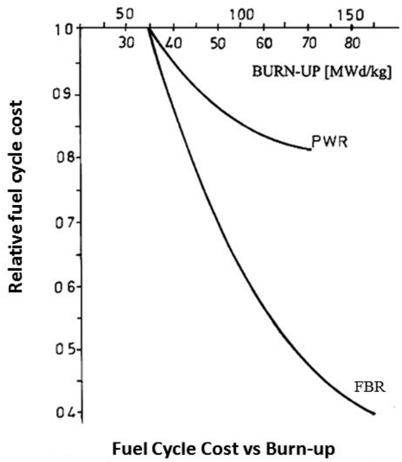


FIG. 1. Fuel Cycle Cost vs Burn-up.

on the closed fuel cycle, India has taken a lead in realizing the technology of liquid sodium cooled fast breeder reactors (FBRs) that operate at a high burnup more than 100 000 MW·d/t. The FBR technology also favours a high margin of fuel cycle cost benefits compared to PWRs (Fig.1). Additionally, an FBR favours reduced generation of high gamma energy nuclear waste because the fuel replacement load is much less due to high burnup compared to a PHWR or PWR.

After gaining long valuable operating experience over few decades with (U, Pu) carbide fuelled Fast Breeder Test Reactor (FBTR), India has built up a (U, Pu) oxide fuelled 500 MW(e) Prototype Fast Breeder Reactor (PFBR) which is in advanced stage of erection and commissioning. The Pu recovered from reprocessing PHWR spent fuel is utilized in the PFBR for the breeding of fissile isotopes, U^{233} from Th^{232} and Pu^{239} from U^{238}, exactly the way H.J. Bhabha, the father of Indian nuclear power, envisioned decades ago.

The processes for fabrication of mixed carbide or mixed oxide are more complicated and challenging than for uranium oxide fuel in terms of metallurgy and radioactive hazard, the selection of material, metallurgy and fabrication of core structurals of FBRs is equally challenging, compared with the zircaloy structurals of thermal power reactors in terms of radiation damage of the material such as irradiation creep, void swelling and irradiation embrittlement, in addition to corrosion due to liquid sodium coolant.

The core structural components comprise fuel subassemblies and hexagonal wrapper tubes (Fig. 2). The NFC has been manufacturing and supplying all the

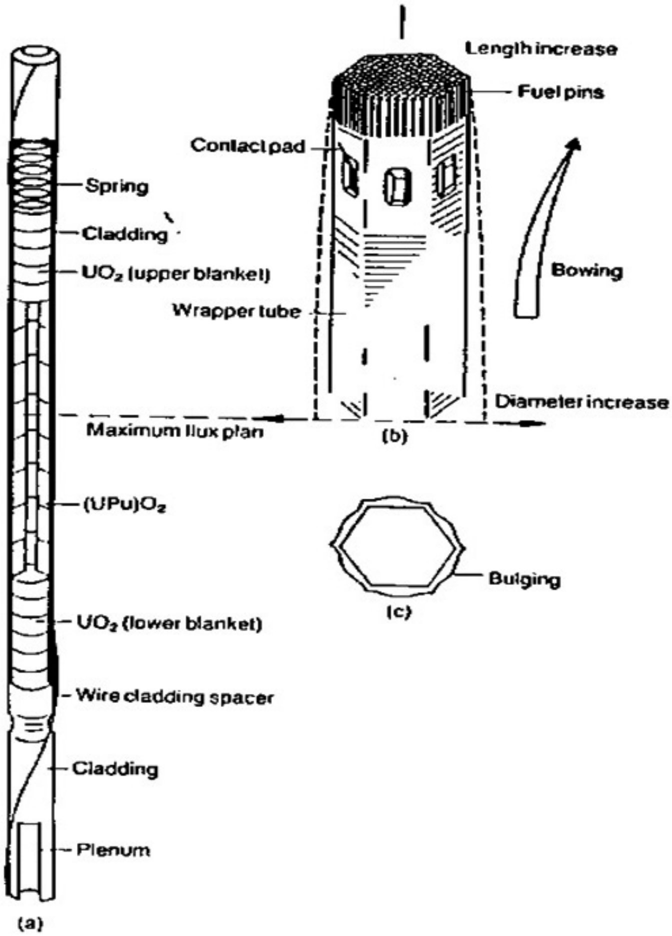


FIG. 2. Irradiation induced deformation of core components.

core subassemblies for FBRs, except the fuel fabrication. The NFC has also supplied special subassemblies for irradiation tests for material characterization and performance testing for the new types of fuel planned for the fast reactors. The NFC has been playing a key role in developing advanced core structural materials through high technology extrusion, cold rolling, innovative annealing cycles, material testing and evaluation.

In the PFBR design, the clad tube experiences a temperature in the range 673–973 K under steady state operating conditions. Under transient conditions, the temperature can rise to 1273 K. Alloy SS D9 in 20% cold worked condition has been chosen for fuel clad and hexagonal wrapper tube for the initial core of the PFBR. The chemical composition of alloy D9 is a modified version of that of alloy SS316. Addition of controlled amounts of Si and Ti, increase of Ni content and lowering of Cr content have been done to SS316 composition for restricting irradiation swelling and irradiation creep behaviour. The hexagonal wrapper tube operates at relatively lower temperatures than the clad tubes. The typical operating temperature of the wrapper tube is 673–873 K which incidentally falls within the peak swelling temperature range. During transients, the temperature may rise up to 1073 K. The peak neutron dose is about 85 dpa similar to clad tube at a burnup of 100 000 MW·d/t.

The major loads experienced by the fuel clad are originating from the internal pressure of accumulated fission gases released from the fuel (about 5 MPa) and fuel clad interaction due to swelling of the fuel. The major loads on the wrapper tube are the internal sodium coolant pressure of about 0.6 MPa and the localized interaction loads at the contact pads owing to bowing of the subassemblies under thermal and swelling gradients, which are depicted in Fig. 2. The typical operating condition of the 500 MW(e) Indian PFBR is given in Table 1.

TABLE 1. TYPICAL OPERATING CONDITIONS OF THE INDIAN 500 MW(E) FBR

Density of neutron flux	$3 \times 10^{15} - 10^{16}$ n/cm^2/s
Rate of He generation	20–30 appm/year
Dose rate	100–200 dpa/year
Temperature	400–650°C

TABLE 2. SELECTION CRITERIA FOR CLAD AND WRAPPER TUBE OF FBR

Criterion	Clad tube	Wrapper tube
Irradiation effects	Void swelling Irradiation creep Irradiation embrittlement	Void swelling Irradiation creep Irradiation embrittlement
Mechanical properties	Tensile strength Tensile ductility Creep strength Creep ductility	Tensile strength Tensile ductility
Corrosion properties	Compatibility with sodium Compatibility with fuel Compatibility with fission products	Compatibility with sodium
	Workability of the material	

The performance of an FBR is largely dependent on the performance of the core structural material i.e. the material of clad tubes and the wrapper tubes under intense neutron irradiation at high neutron flux ($\sim 10^{15}$ n/cm^2/s). The major criteria for selection of materials for clad and wrapper tube is given in Table 2. These components undergo dimensional and strength degradation due to void swelling, irradiation creep and irradiation induced helium embrittlement, which determines the permissible service life of the fuel subassemblies in the reactor core. Neutron irradiation causes formation of vacancies and other point defects. As per thermodynamic principles, these vacancies condense into clusters in the form of void nuclei. The void nuclei become stabilized in the presence of He formed due to (n,α) reaction. The void can grow by diffusion of vacancies or by coalescence with other voids resulting in swelling. Even at low concentrations, helium gas can have life limiting consequences for materials because of its low solubility in the crystal lattice. The gas accumulates at defects, dislocations and grain boundaries, leading to swelling or embrittlement. The mechanism of void swelling, embrittlement and irradiation creep is depicted in Fig. 3. Therefore, the selection of material and manufacturing route for clad and wrapper tube has evolved to minimize radiation damage under different neutron flux and the operating temperature prevailing in the FBR. With increase of neutron dose on material, the core structural material in austenitic grades has been progressively modified from CW316 to CW316Ti to D9 and D9I (Fig. 4) for radiation doses up

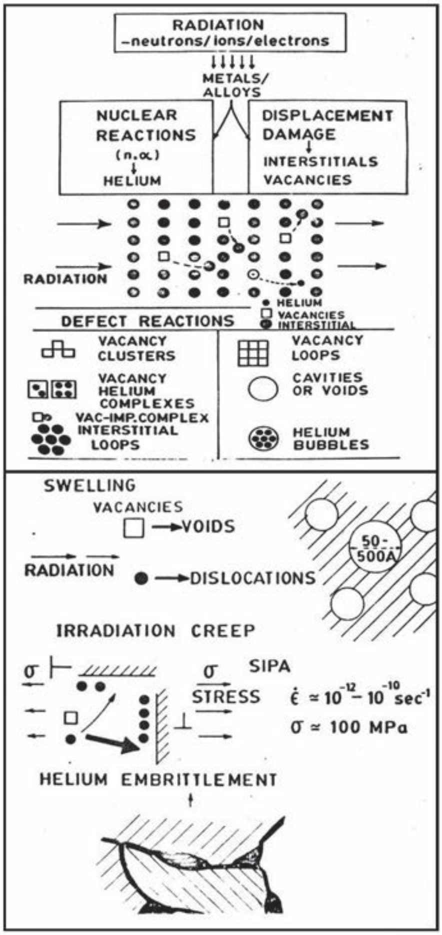


FIG. 3. Mechanism of irradiation.

to about 140 dpa and an operating temperature prevalent with oxide fuel. Though austenitic stainless steel has excellent high temperature mechanical properties, excellent formability and compatibility with liquid sodium, the material is not recommended for use in high burnup situations owing to inadequate void swelling resistance at higher radiation doses.

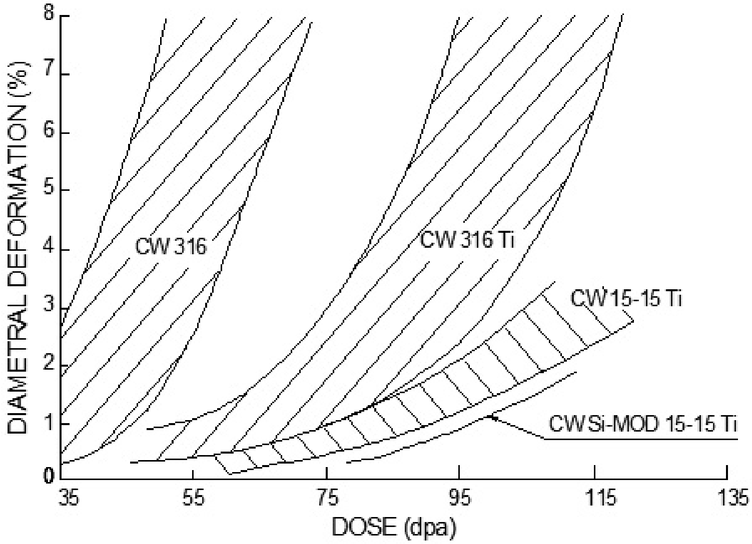


FIG. 4. Selection of material as per neutron dose.

The 9-12% Cr ferritic-martensitic steels are considered as the long term solution for FBR core structural materials. Although these alloys (9Cr-1Mo (EM10), Modified 9Cr-1Mo (Gr. 91), 9Cr-2MoVNb (EM12), 12Cr-1MoVW (HT9), etc.) have excellent swelling resistance to doses even up to 200 dpa, (1% swelling reported in HT9 after irradiation at 693 K at 200 dpa), their thermal creep resistance decreases drastically above 823 K. Therefore, they are not suitable for clad tubes. High thermal creep strength is not a primary requirement for the wrapper material since it operates at lower temperatures. Reduced creep strength is therefore acceptable. However, the increase in ductile to brittle transition temperature (DBTT) due to irradiation is a cause of concern for ferritic steels. Consequently, extensive studies involving modification of the composition and initial heat treatments have been carried out to improve the fracture toughness. The upper shelf energy and shift in DBTT (Fig. 5) appear to saturate at high irradiation doses. Significant increase in toughness (i.e. low DBTT and high upper shelf energy) had been realized by (i) avoiding the formation of delta-ferrite (which favoured void swelling) by ensuring a fully martensitic structure in 9-12% Cr steels and (ii) optimizing the austenitizing temperature to obtain fine grained martensitic structure and tempering treatments to reduce the strength of the martensite. Modified 9Cr-1Mo grades of ferritic steels are preferred for wrapper tube application.

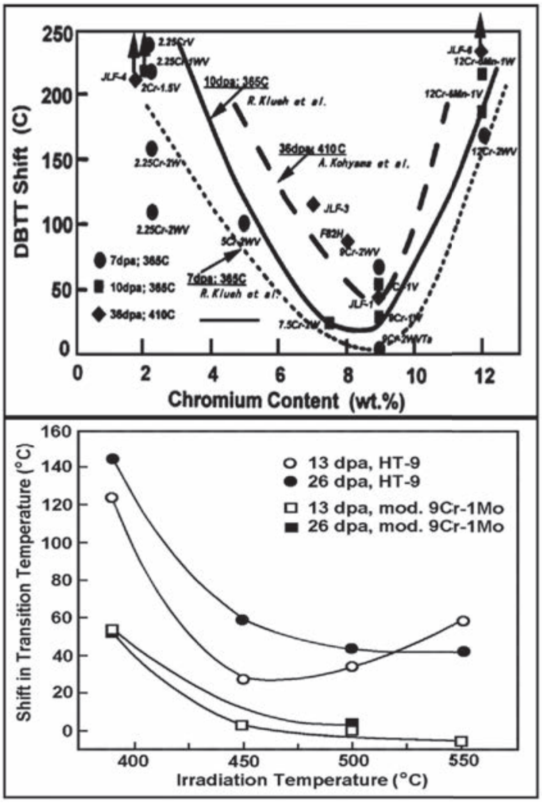


FIG. 5. DBDT temperature of T91 steel.

Oxide dispersion strengthened (ODS) steel is preferred for operating the reactor at still higher fluence to derive higher burnup. The ferritic-martensitic grade 9Cr-ODS dispersed with 0.35% Y_2O_3 has been successfully developed and manufactured at the NFC for irradiation test. The material combines the swelling resistance of Modified 9Cr-1Mo and the thermal creep properties of SS D9, hence it is ideal for longer service life of the mixed oxide fuel subassemblies. However, workability of the material in rolling is an issue which is taken care of by proper design of thermomechanical processes. The material is required to be produced through mechanical alloying under inert atmosphere to avoid oxidation of submicron particles. Hence, manufacturing of this ODS alloy involves additional process steps such as production of constituent element powder through atomization, mechanical alloying by high energy ball milling under a protective atmosphere, vacuumized canning, upsetting followed by extrusion and rolling operations.

Sustainability of nuclear power is one of the important issues. It is mandatory to increase the breeding ratio to sustain the growth of nuclear energy, hence metallic fuel is desirable. Accordingly, alloys of U-Pu-Zr are chosen for India's future metallic fuelled fast reactors. The choice of clad and wrapper materials to be used with these fuels will be based on the fuel–clad mechanical and chemical interactions. However, owing to higher swelling, inferior creep strength above 820 K and inferior compatibility of the metallic fuel with clad tube, clad materials such as SS D9 & T91 are not acceptable. Uranium forms a low melting point eutectic with Fe at 725°C (Fig. 6), which may be reached during transient operating conditions and other unusual occurrences. Therefore, double clad tube, i.e. T91 tube lined with Zr-alloy tube, is being developed at the NFC. Zirconium has been chosen as an inner liner because it is isomorphous with uranium (Fig. 7) and has excellent corrosion resistance. The co-pilgering route is followed to produce this double clad tube. The microstructure evaluation showed excellent bonding between the layers.

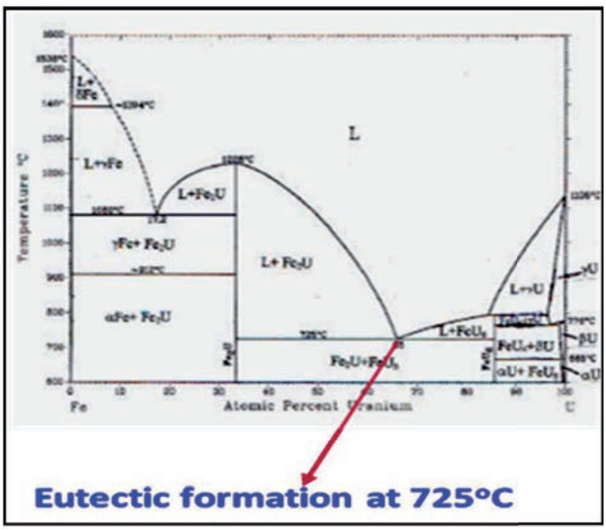


FIG. 6. Phase diagram of Fe and U.

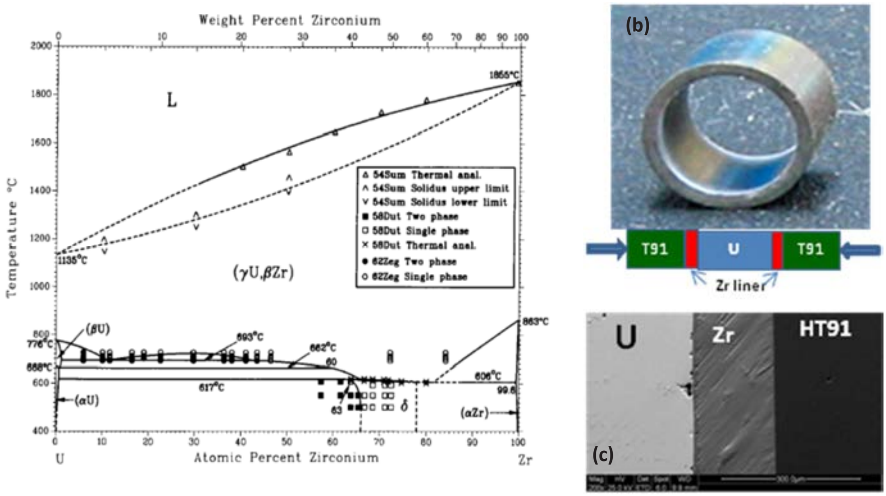


FIG. 7. (a) Phase diagram of U and Zr, (b) Zr-4 lined T91 clad tube, (c) Diffusion couple of U, Zr-4 and T91.

2. CRITICAL ASPECTS OF MANUFACTURING CLAD TUBES AND WRAPPER TUBES

2.1. Methodology adopted

Since the product is being manufactured for the first time in India, an integrated approach was followed. The critical requirements of the finished tubes were thoroughly studied and analysed by the key agencies, i.e. designer, raw material manufacturer, tube producer, QA and QS, and the end user to arrive at meaningful process development. Initially, feasibility studies were made followed by pilot production for qualification. The problems encountered during pilot production were solved before start of actual production.

2.2. Process

A typical process flowsheet followed is:

forged rods cutting
↓
billet machining
↓
hot extrusion
↓
deglassing
↓
ID honing and OD grinding
↓
ultrasonic testing
↓
cold rolling
↓
heat treatment
↓
cold rolling/drawing
↓
inspection and testing

3. MICROSTRUCTURE AND MECHANICAL PROPERTIES OF THE D9 CLAD TUBE, WRAPPER TUBE AND MODIFIED 9CR-1MO CLAD TUBE MANUFACTURED AT NFC (see Figs 8–11 and Tables 3–5)

3.1. Mechanical properties of D9 clad and wrapper tube

TABLE 3. MECHANICAL PROPERTIES D9 CLAD TUBE (TYPICAL TEST RESULTS)

Properties	Room temp.	Typical results obtained
UTS (MPA)	700–830	791
YS (MPA)	550–760	709
% Elg.(GL=5.65√A)	20 min.	20
Hardness (VHN) max. (0.5 kgf load)	220–290	230–233
UTS (MPA)	500–690	559
YS (MPA)	430–580	507
% Elg.(GL=5.65√A)	10 min.	10

3.2. Microstructure and mechanical properties of hexagonal wrapper tube

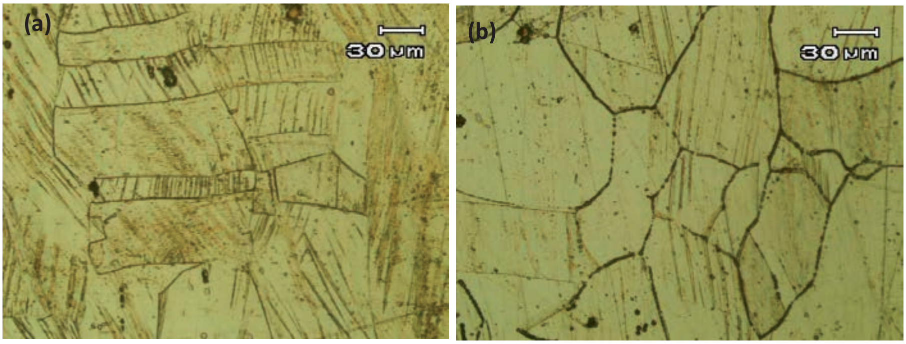


FIG. 8. (a) Microstructure of 20% cold worked wrapper tube, (b) microstructure of wrapper tube after sensitization at 650°C for 1 hour showing dual structure.

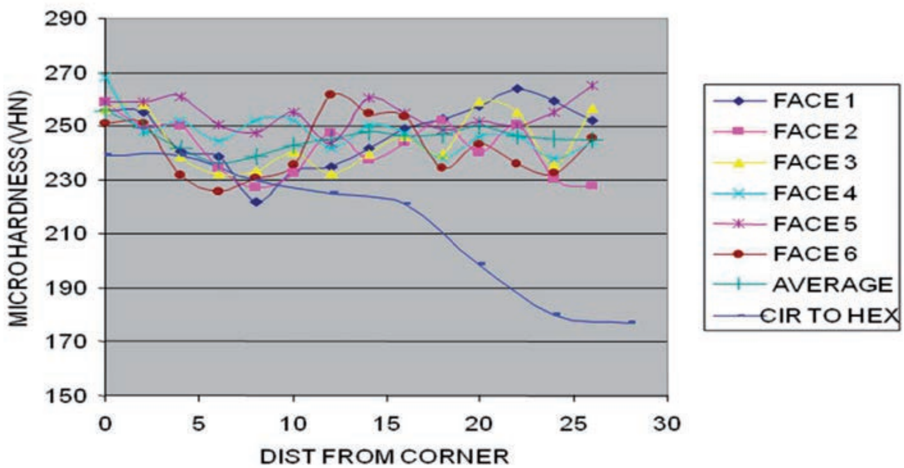


FIG. 9. Variation of micro hardness across the face of hexagonal wrapper, manufactured by hexcan to hexcan and circular to hexcan cold rolling.

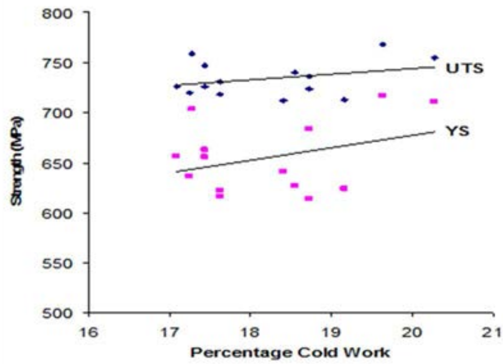


FIG. 10. Variation of UTS and YS with % cold work in D9 clad and wrapper tube.

3.3. Metallurgical and mechanical properties of modified 9Cr-1Mo clad tubes

TABLE 4. MECHANICAL PROPERTIES OF CLAD TUBES

Properties	Room temp.	As manufactured	As simulated heat treated
UTS (MPA)	585–760	695–714	660–710
YS(MPA)	420 min.	535–578	470–520
% Elg.(GL=2”)	20 min.	23–26	24–26
Hardness	250 max.	233–248	
	At 540°C		Results obtained
UTS (MPA)	370 min.		465–525
YS(MPA)	275 min.		370–420
% Elg.(GL=2”)			20–22

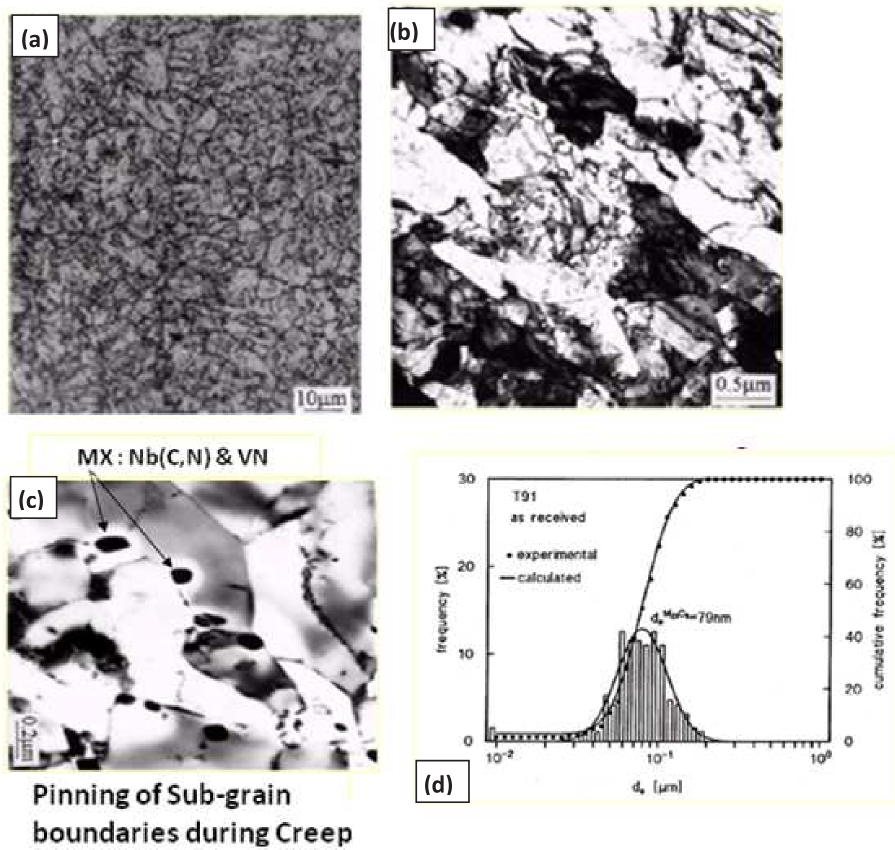


FIG. 11. (a) Normal, (b) magnified microstructure of Mod.9Cr-1Mo clad tube showing tempered martensite structure (c) high resolution TEM micrographs indicates the presence of niobium carbonitrides, i.e Nb(C, N) and vanadium nitrides ppts predominantly along the subgrain boundaries. These ppts are responsible for its improved creep and high temperature strength, (d) average precipitation size and distribution indicates that average size of ppts are 80 nm.

3.4. Specifications requirements of the D9 and modified 9Cr-1Mo(T91) clad tubes

TABLE 5. CHEMICAL COMPOSITION OF D9 AND MODIFIED 9CR-1MO (T91)

Chemical composition of D9				Chemical composition of modified 9Cr-1Mo(T91)			
Element	wt%	Element	wt%	Element	wt%	Element	wt%
C	0.035–0.05	Ta	0.02 max	C	0.08–0.12	P	0.02 max
Cr	13.5–14.5	Ti	5–7.5	Mn	0.3–0.5	Ni	0.2 max
Ni	14.5–15.5	Al	0.05 max	Si	0.2–0.5	S	0.01 max
Mo	2–2.5	Si	0.5–0.75	Cr	8.0–9.0	Ti	0.01 max
Mn	1.65–2.35	Co	0.05 max	Mo	0.85–1.05	Cu	0.1 max
N	0.005 max	Cu	0.04 max	N	0.03–0.07	Sn	0.02 max
S	0.01 max	As	0.03 max	Nb	0.06–0.1	Al	0.04 max
P	0.02 max	V	0.04 max	V	0.18–0.25	Sb	0.01 max
B	10–20 ppm	Nb	0.05 max	Fe	Balance		
Fe	balance						

— **Dimensional tolerances:**
  - OD: 6.6 ± 0.02 mm
  - ID: 5.7 ± 0.02 mm
  - T: 0.43 min
  - Length: 2555 + 1/-0 mm
  - Straightness: 0.25/500

— **Cold work:** 20 + 4 % for D9 only, no retained cold for T91

— **Grain size** after final annealing: 7-9 (ASTM E-112) for D9

— **IGC:** ASTM A 262 Pr A @ 100 X magnification

— **Hardness :** 220-290 VHN for D9 & 250 VHN max for Modified 9Cr-1Mo

— **Surface finish:** 0.5 μRa

— **ECT:** 0.3 ± 0.03 mm through holes
— **UT:** 0.05 ± 0.002 mm X 1.5 mm Lg 60° V Notch

4. MANUFACTURING CHALLENGES FOR ODS CLAD TUBE

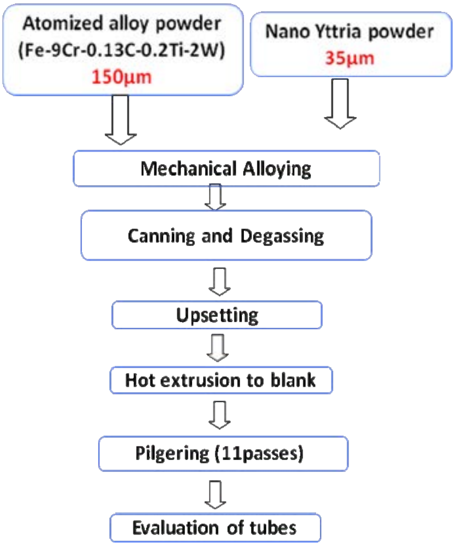
ODS alloy was made through mechanical alloying as per the following composition (Table 6) and general flow sheet given below:

The manufacturing challenges are given in the following subsections.

TABLE 6. CHEMICAL COMPOSITION OF THE ODS ALLOY STEEL

Element	wt%	Element	wt%
C	0.12	Y_2O_3	0.36
Cr	8.85	N	0.01
W	2.01	Mn	0.01
Ti	0.21	O (Total)	0.12
Fe	Balance		

General Flow Sheet



4.1. Mechanical alloying

A master alloy of Fe-9Cr-0.13C-0.2Ti-2W was used for manufacturing atomized alloy powder. The average particle size of the alloy powder is around 300 μm and Y_2O_3 particle size of around 35 μm. ODS alloy powder is milled in a high energy ball mill for 4 hours to achieve the desired particle sizes. Final powder obtained has a plate-like morphology and average particle size of the Y_2O_3 is 6–8 nm (Fig.12). The alloy powder was canned in mild steel can followed by vacuum degassing at 350°C for 3 hours. Degassed alloy powder filled mild steel can is used for upsetting to get desired density.

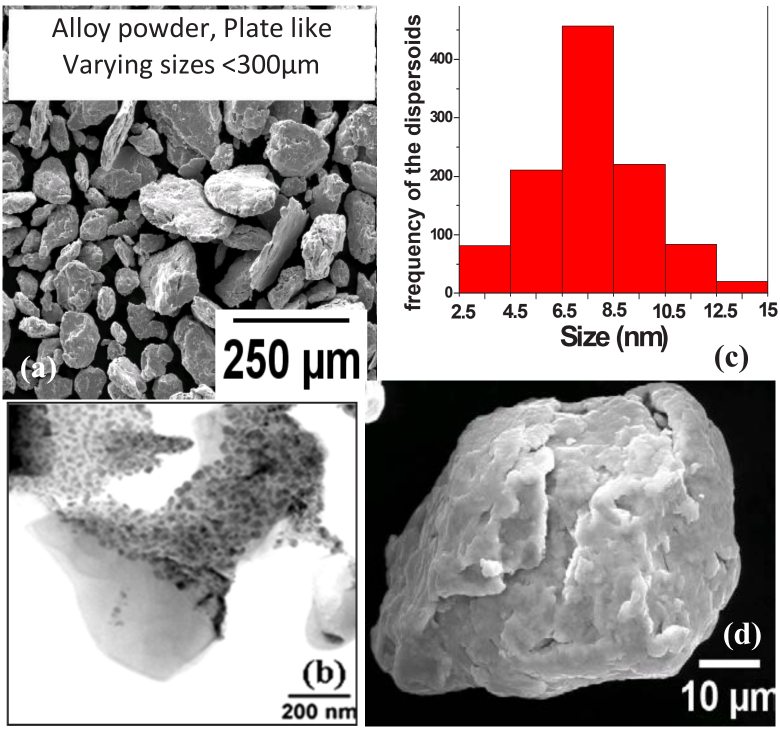


FIG. 12. (a) Alloy powder plate like morphology, (b) and (c) powder morphology of alloy powder, (d) yttria particle size and their distribution after mechanical alloying.

4.2. Consolidation

— Reactivity of powder is very high. Hence powder-can should not puncture during upsetting. This was achieved through:
  - Optimized canning thickness;
  - Reduced clearance between canned billet and container;
  - Slow ram speed in during upsetting.
— Optimization of upsetting temperature, force and time.
— Development of tools for upsetting at 630T vertical extrusion press.

4.3. Extrusion

— Optimization of extrusion temperature for high strength of base material and lower strength of can (MS).
— Optimization of extrusion speed: slow speed is required for homogeneous deformation but also results in loss of heat.
— Selection of lubrication.

4.4. Removal of MS patches

Detection and removal of MS patches was extremely challenging. An innovative NDT process (Fig. 13) was developed for detection of MS patches and removed by local conditioning.

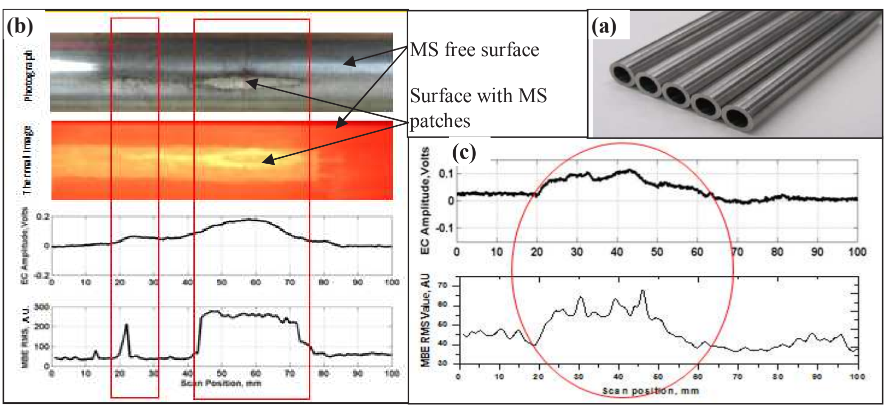


FIG. 13. (a) Machined blank of size 36 mm OD × 4 mm WT, (b) indication of presence of MS patches on the blank surface in different methods, (c) MS patches detected by EC and MBR techniques.

4.5. Mechanical and metallurgical properties of 9Cr–ODS clad tube

TABLE 7. MECHANICAL PROPERTIES OF THE ODS CLAD TUBE

Comparison of mechanical properties				
	Room temp.		At 540°C	
Material	D9	ODS	D9	ODS
UTS (MPa)	724	921	548	554
YS (MPa)	600	830	488	519
% Elong.	21	22.7	14	14

The mechanical properties achieved are superior to the mechanical properties of D9 clad tube (Table 7). In Fig. 14(a), a TEM study reveals the presence of carbide at the subgrain boundary. Average particle sizes of dispersed Y_2O_3 are 6 nm and are distributed uniformly throughout the matrix (Fig.14(b)). EBSD image (Fig.14(c)) of the finished tube sample confirms absence of any anisotropy. Creep rupture data (Fig.14(d)) indicate it is equal or superior to D9 clad tube at higher temperatures. Mechanical properties of the finished clad tube developed at NFC indicates that the strength is comparable to the values reported in the literature.

5. DEVELOPMENT OF BI-METALLIC CLAD TUBE FOR METALLIC FUEL

Sustainability of nuclear power is one of the important issues, therefore it is mandatory to reduce doubling time. Hence, metallic fuel is the definite choice owing to its high breeding ratio. For a metallic fuelled FBR, U-Pu-Zr is the choice as fuel, but selection of clad material for the metallic fuel is a challenging task. Iron based alloys are not compatible with metallic fuel owing to the formation of low melting eutectic at 725°C (Fig. 6). Therefore, modified 9Cr-1Mo (T91) alone is not suitable for clad tube; even though a metallic fuelled FBR operates at a low temperature and T91steel has excellent void swelling resistance. The concept of a bi-metallic clad tube in which the liner will be compatible with uranium and the outer clad tube will have excellent void swelling resistance was developed.

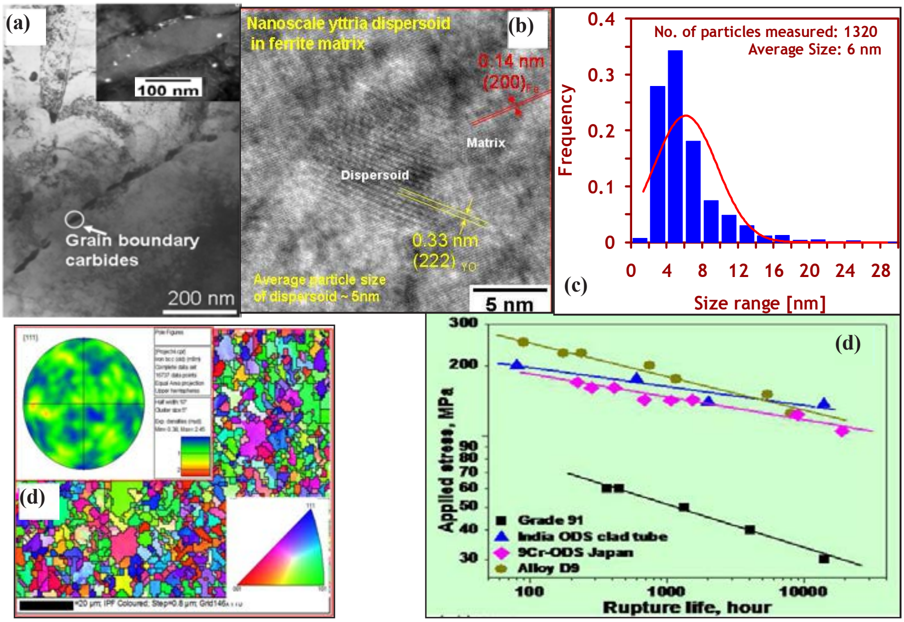


FIG. 14. *(a) TEM micrograph indicating presence of grain boundary carbide, (b) yttria particle size and distribution, (c) EBSD data, (d) creep rupture data.*

Zirconium can be used as an inner liner as it is isomorphous with uranium and the outer clad is T91 steel, which have excellent void swelling resistance.

5.1. Fabrication methods

Manufacturing of T91 clad tube with Zr alloy lining is a challenging task owing to different deformation characteristics and physical properties of Zr alloy and T91 steel. The short listed methods of manufacturing of bi-metallic clad tubs are:

- Co-extrusion;
- Drawing;
- Internal expansion;
- Explosive bonding;
- Co-pilgering.

5.2. Co-Pilgering technique for manufacture of T91 clad tube with Zr alloy lining

The co-pilgering route was chosen over other routes for the manufacturing of T91 clad tube with a Zr alloy lining because pilgering is an excellent technique of tube reduction, in which high compressive stresses act on the outer and inner surfaces of the tube to reduce the diameter and wall thickness. In the case of liner clad, these forces affect the thickness reduction of both clad and inner liner simultaneously. Hence effective mechanical bonding can be ensured. A typical flowsheet for manufacturing double clad tubes is given in Fig. 15.

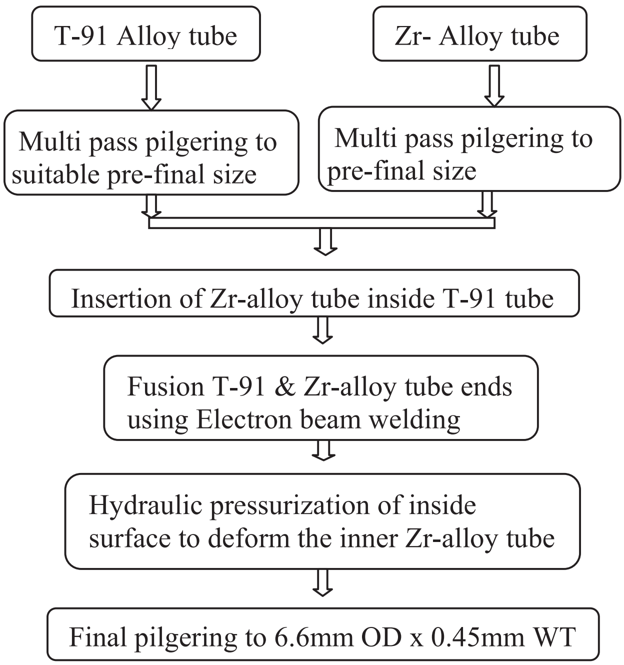


FIG. 15. Typical flowsheet for manufacturing double clad tubes.

5.3. Challenges of co-pilgering of T91 clad tube with Zr-alloy liner

Both materials have different deformation mechanisms and differing physical properties such as coefficient of linear expansion. The temperature of phase change of these two different materials are also critical from a manufacturing point of view. The major challenges of co-pilgering are:

- T91 and Zr alloys show different behaviour during the deformation and heat treatment process owing to their different material characteristics. Hence, in designing the pilgering pass schedules of T91 & Zr, alloy tube is important to obtain the required dimensions after co-pilgering.
- Since the pilgering process uses oil for lubrication, there is a chance of oil seepage between the layers. To overcome this:
  (i) A mathematical model was developed to find the parameters for the input to equate the strains of the two materials during co-pilgering.
  (ii) End fusion of the liner and clad was done to reduce the relative slip and prevent ingress of oil between two layers.

5.4. Evaluation of the double clad tube

The main criteria for qualification of the double clad tube is to check the bond and the gap between two layers by:

- Destructive testing;
- Non-destructive testing.

Destructive testing followed by SEM gives the cross-sectional view of the double clad tube, which shows the bond between two layers and the joining line of the two layers. For qualification testing of the entire length, non-destructive testing was used.

The clad tube was loaded with metallic fuel and exposed to 700°C for 1505 hours and then analysed by EPMA scan, as shown in Fig. 16. T91, Zr, U strongly bonded at interfaces and there is no interfacial gap between layers. This ensures that there is no resistance to heat transfer during reactor operation.

6. MANUFACTURE OF PFBR FUEL SUBASSEMBLIES

NFC manufactures and supplies the hardware such as fuel clad tubes and all the other pin components for the manufacture of the MOX fuel pin. The fuel pin components are shown in Fig. 17 and the material used for their construction is given in Table 8.

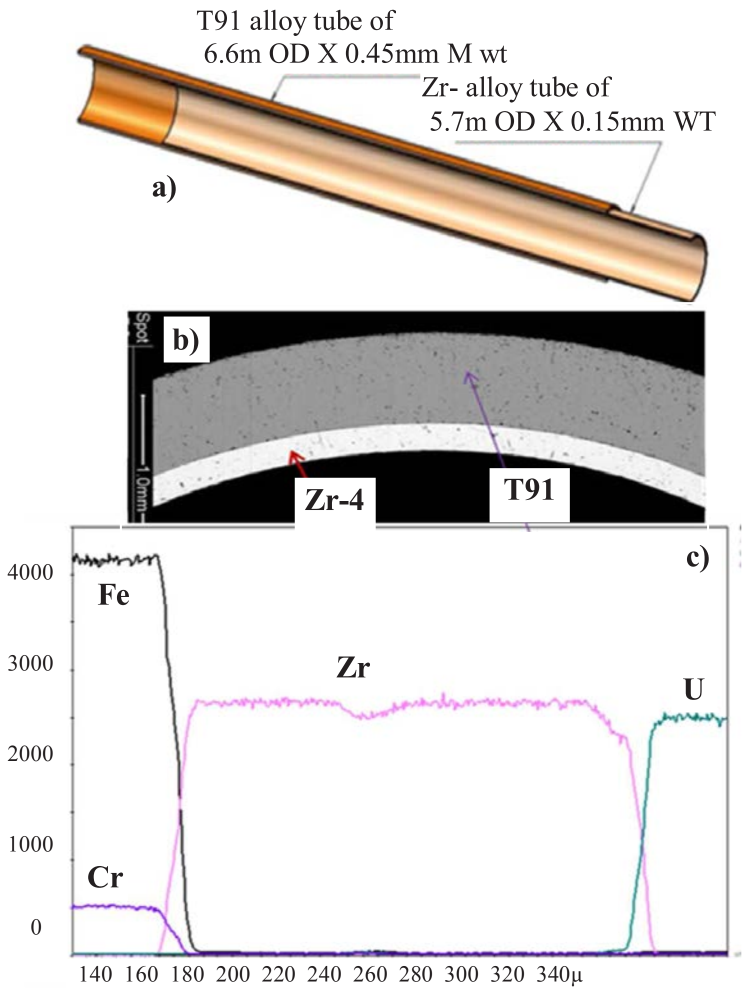


FIG. 16. (a) Model of double clad tube, (b) SEM image of T91 clad tube with Zr-4 liner, (c) EPMA scan along U/Zr/T91 interface.

FIG. 17. MOX fuel pin components.

TABLE 8. PIN COMPONENTS WITH MATERIAL OF CONSTRUCTION

Sl. No.	Component	Material
1	Clad tube (crimped)	D9
2	Top plug	SS 316LN
3	Bottom plug	SS 316LN
4	Middle plug	SS 316LN
5	Spring	ASTM A453 grade 660
6	Spring support	SS 316LN
7	Spacer wire (with bead forming)	D9

Some 217 sets of the above components are required for the manufacture of one fuel subassembly and a total of 197 fuel subassemblies are required for the first PFBR core. All the above components have intricate profiles requiring a high degree of accuracy in dimensional and geometrical measurements.

The MOX fuel pellets and axial blanket pellets are manufactured at a separate facility and they are finally encapsulated with the help of the pin hardware to make the fuel pins. These fuel pins are loaded into pin magazines and assembled along with other hardware to make the pin bundle. Subsequently, the hexagonal sheath is assembled over the pin bundle and the complete circumferential welding of the hexagonal tube is done with the lower part assembly to make the complete subassembly.

Other than the pin components, the fuel subassembly has numerous components having intricate shapes and stringent dimensional and geometric tolerances. Machining of these components poses a major challenge owing to the ductile nature of the material. Pictures of a few of the fuel subassembly components/part assemblies are shown Fig. 18.

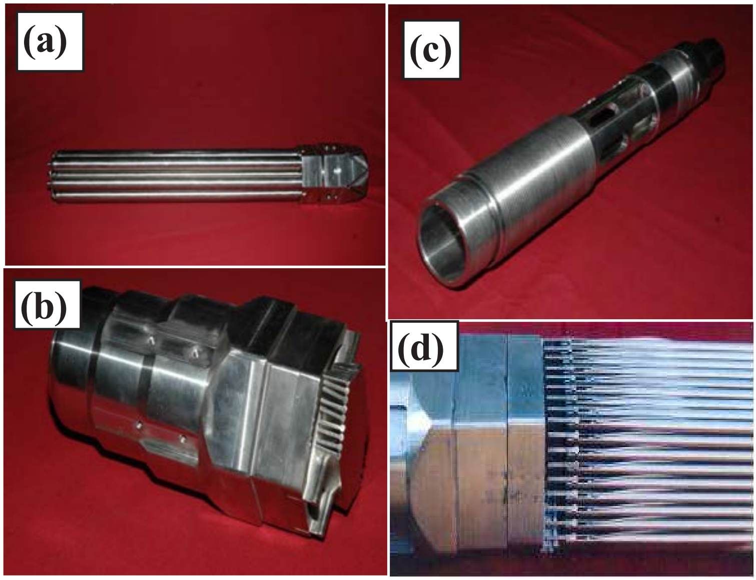


FIG. 18. (a) Handling head assembled with top axial shielding pins, (b) foot upper part assembled with fuel pin support assembly and guide rails, (c) coolant entry tube with discriminator, (d) fuel pin bundle after assembling.

For the final assembly of the fuel pins and other subassembly components, a manual fuel pin assembly fixture was made. Special purpose machines, such as a TIG welding system for automatic hexagonal sheath welding, the spacer wire wrapping and spot welding system, the bottom end plug welding system, the automatic crimping machine and button forming machine, etc., were developed for manufacturing the fuel subassembly of the PFBR. The manufacturing flowsheet for the PFBR fuel subassembly is depicted in Fig. 19.

7. CONCLUSION

India is relying on growth of nuclear power to foster the energy security of the nation. Development of fast breeder technology is proceeding at a greater pace in the country to achieve power generation through plutonium and breeding of U^{233} from Th^{232} and Pu^{239} from U^{238}.

Materials for core structural components, including fuel clad and wrapper tubes, face service life limitations because of the radiation damage through void swelling, irradiation creep and irradiation embrittlement under high fluence.

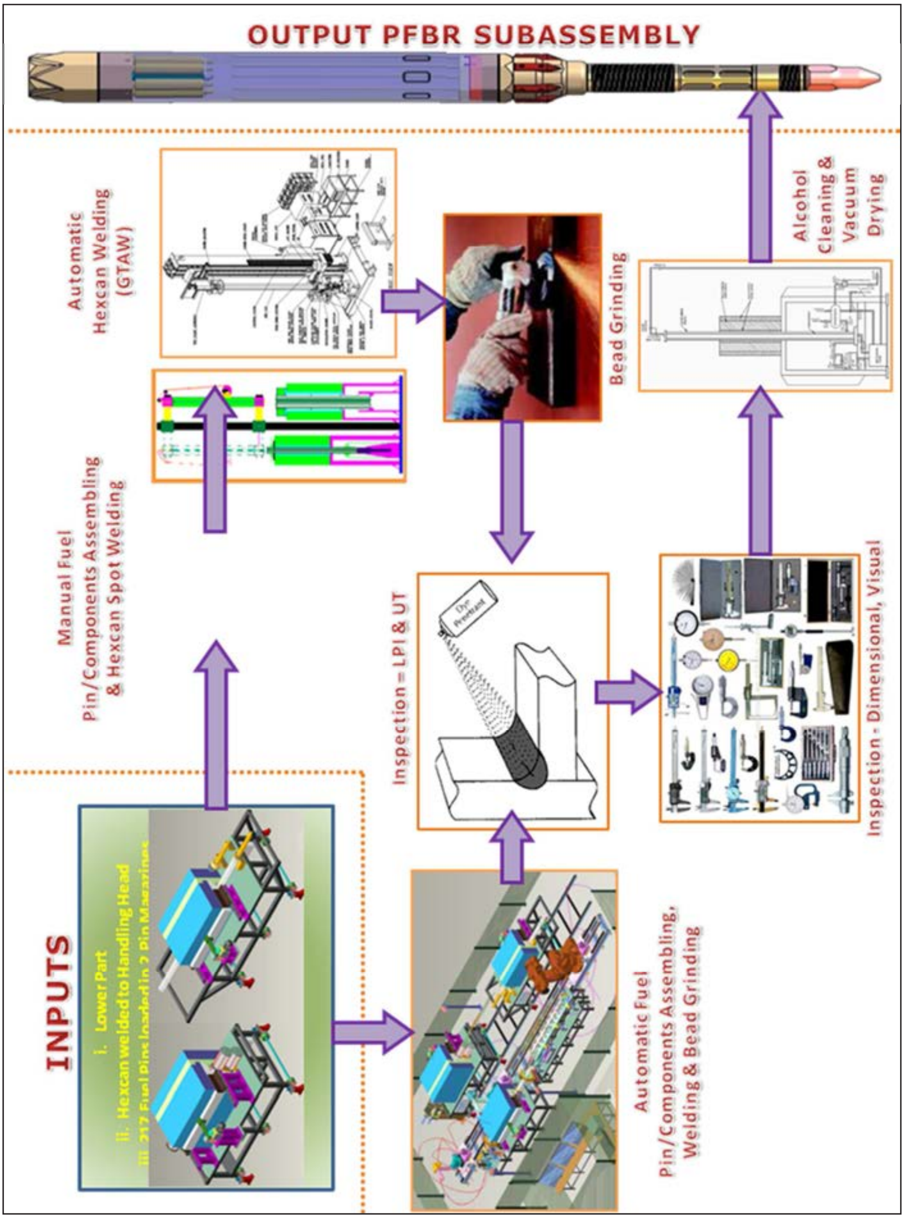


FIG. 19. Manufacturing flow sheet for PFBR subassembly.

The NFC is a responsible partner in developing high performance materials and is solely responsible for establishing the manufacturing processes for the core structural components of the breeder reactors. The NFC has successfully manufactured clad tubes and hexagonal wrapper tubes made in austenitic grade SS D9 material for the first PFBR being installed in the country.

Development has been completed for advanced future materials such as austenitic steel SS D9I for fuel clad tubes, 9Cr-1Mo (T9) ferritic steel for wrapper tubes, 9Cr-ODS, i.e. Y2O3 dispersion strengthened ferritic-martensitic fuel clad tube for higher burnup with mixed oxide fuel, zirconium lined modified 9Cr-1Mo clad tube for future metallic fuel, etc., to achieve higher burnup with reduced doubling time.

The NFC has also manufactured high precision fuel pin components and subassemblies by using indigenously developed equipment, such as an end plug welding machine, spacer wire wrapping and spot welding equipment, a tube crimping machine, a button forming machine and automatic hexagonal sheath welding equipment.

BIBLIOGRAPHY

BALDEV, R., MANNAN, S.L., VASUDEVA RAO, P.R. and MATHEW, M.D., Development of fuel and structural materials for fast breeder reactors", Vol. 27, Pt 5, Indira Gandhi Centre for Atomic Research, Kalpakkam, India (2002) 527–558.

CHETAL, S.C., BALASUBRAMANIYAN, V., CHELLAPANDI, P., MOHANAKRISHNAN, P., PUTHIYA VINAYAGAM, P., PILLAI, C.S., et al., The design of the Prototype Fast Breeder Reactor, J. Nucl. Eng. Des. **236** (2006) 852–860.

GILBERT, M.R., DUDAREV, S.L., ZHENG, S., PACKER, L.W., SUBLET, J.C., Transmutation, gas production and helium embrittlement in material under neutron irradiation, Rep. CCFE-PR(12)02, EURATOM/CCFE Fusion Association, Culham Science Center Abingdon, Oxfordshire, UK (2012).

HOFMAN, G.L., WALTERS, L.C., BAUER, T.H., Metallic fast reactor fuels, J. Prog. Nucl. Energy, **31** 1–2 (1997) 83–110.

MANNAN, S.L., CHETAL, S.C., BALDEV, R., BHOJE, S.B., Selection of materials for prototype fast breeder reactor, Trans IIM, **56** (2003) 155.

MATERIALS PERFORMANCE IN SODIUM COOLED FAST REACTORS: PAST, PRESENT AND FUTURE

K. NATESAN, M. LI
Argonne National Laboratory,
Argonne, United States of America

Abstract

This paper gives an overview of the requirements, selection, and performance of materials for in-core and out-of-core components in a sodium cooled fast reactor. The potential degradation/damage mechanisms are identified and key variables controlling the damage are enumerated. Historically, austenitic stainless steels have been used for the in-core components such as fuel cladding, ducts and other internals. The major concern with these steels is the neutron radiation damage that can lead to swelling and dimensional changes and loss of ductility. In the case of out-of-core components, again austentic stainless steels have been used for the primary and intermediate heat transport circuits that include primary and secondary piping, IHX, etc., in past reactors. Low alloy steels and austenitic stainless steels have been used for the evaporators and steam superheaters, respectively, in past reactors. Advanced materials are a critical element in the development of sodium reactor technologies. Enhanced materials performance can lead to increased efficiency with higher sodium reator outlet temperatures, can improve safety margins, and can provide design flexibility, but also is essential for the economics of future advanced sodium reactors. In this regard, research is being conducted to develop more radiation tolerant, high strength materials such as ferritic-martensitic alloys, including oxide dispersion strengthened steels for in-core applications, a class of 9Cr-12Cr steels for primary and secondary piping, and 9Cr steels for the steam generators for application in future reactors.

1. BACKGROUND

The objectives of sodium cooled fast reactors (SFRs) are to expand the use of nuclear energy to meet the increasing global energy demand and to address nuclear waste management. The added benefit of nuclear energy is the absence of greenhouse gases in power generation. The primary goals of the ongoing activities in the R&D for SFRs are (1) to demonstrate reactor based transmutation of transuranics, (2) to qualify the transuranics-containing fuels and advanced structural materials needed for a full scale SFR, and (3) to support R&D and demonstration required for certification of an SFR standard design by the regulatory agency. Additional objectives include (1) incorporation and demonstration of innovative design concepts and features, (2) demonstration of improved technologies for safeguards and security, and (3) support for development of design, fabrication and construction, testing and deployment of systems, structures and components infrastructure for the SFR in the United States of America.

A typical SFR can be either a pool or loop type design with the same objective of transferring heat from the core through primary and intermediate sodium circuits to a steam generator and a power turbine. Figure 1 shows a schematic of the two design concepts [1]. Globally over the past 40 years or more, several sodium reactors of various sizes and capacities have been built and operated and substantial databases have been developed on fuel performance, materials selection and performance, and component design and operation. Table 1 lists several reactors, both pool and loop types, built in various countries, some have operated for 30 years or more. The results from these experimental and demonstration reactors have contributed to an understanding of the generic issues as well as system specific issues in sodium reactors.

2. MATERIALS AND KEY COMPONENTS IN SFRS

The major components in an SFR encompass the in-core components such as fuel cladding, duct, wrapper wire, etc., and out-of-core components that can be grouped under the reactor system and primary containment boundary, primary heat transport system, intermediate heat transport system

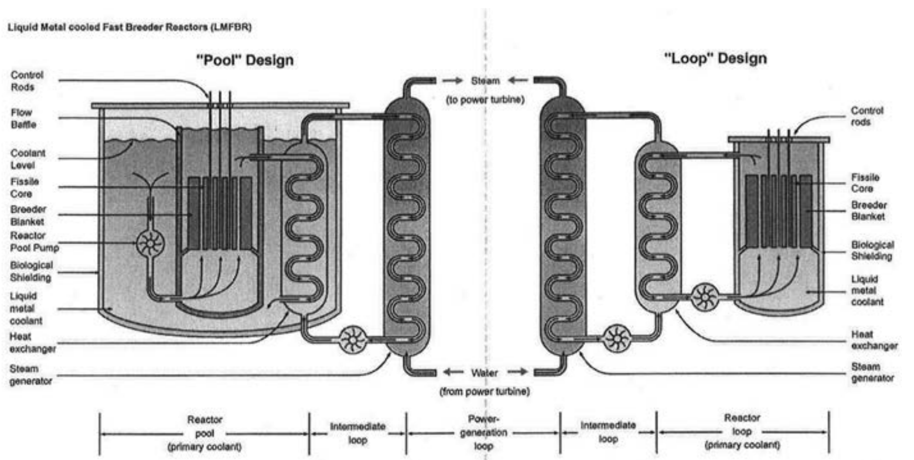


FIG. 1. Schematic of pool and loop designs of SFRs [1].

TABLE 1. A LISTING OF EXPERIMENTAL AND DEMONSTRATION SFRS

Country	Reactor	MW(th)	Reactor type	Operating life (years)	Inlet to IHX temp. (°C)	Inlet to SG temp. (°C)
USA	EBR-II	62.5	Pool	30.9	473	467
	FFTF	400	Loop	12.1	565	[a]
UK	DFR (NaK)	60	Loop	14.4	350	330
	PFR	650	Loop	20.1	550	540
Russian Federation	BOR-60	55	Loop	39.9	545	480
	BN-350	750	Loop	26.4	430	415
	BN-600	1470	Pool	29.7	550	520
France	Phenix	563	Pool	36.2	560	550
	SuperPhenix	2990	Pool	12.4	542	525
Japan	Joyo	140	Loop	32.5	500	470
	Monju	714	Loop	15.6	529	a

[a] No steam generator and only sodium to air heat exchanger.

and power generation system. Over the past 40 years, extensive research and development has been conducted on the irradiation, heat and corrosion resistance of a variety of materials for application in in-core and out-of-core functional and structural components. The large database and lessons learned from operating past reactors can be applied in the design, materials selection and construction of components for current and future SFRs. Furthermore, information relative to the reliability and performance of large sodium pumps, flowmeters, and valves, the thermohydraulic performance of sodium heated steam generators, and requirements for sodium purity control and monitoring equipment is well documented from past operated reactors. The purpose of this paper is to give an overview of the materials that have been used in past reactors, the improvements needed to increase the efficiency of the system and to advance materials technology to enable a reactor system with increased fuel flexibility, increased safety, improved economics, compact design and lower cost.

Table 2 lists the materials that have been used in a number of SFRs for fuel cladding, reactor vessel, primary/secondary piping, intermediate heat exchanger (IHX) and steam generator. In the case of in-core components such as fuel cladding, fuel assembly ducts, wire wrapper, etc., the primary material considerations are resistance to radiation, such as minimal or acceptable void swelling and associated dimensional changes and minimal ductility loss, sodium and fuel compatibility (especially for cladding), and structural integrity over the service period. Based on these requirements, either Type 316 stainless steel or a minor variant of it has been predominantly used for fuel cladding in all the past reactors. The structural materials for the primary and secondary heat transport systems as well as for downstream components, such as steam generators and steam turbines, were selected on the basis of the functional requirements as well as on the basis of ASME Code qualified materials. In the past reactors, the sodium core oulet temperature has been in the range of 500–550°C and the thermal efficiency of the nuclear plant has been relatively low. Figure 2 shows a plot of ideal Carnot thermal efficiency and the efficiencies of various SFRs. While the steam temperatures in SFRs is around 500°C, the coal based fossil systems routinely operate at a steam temperature of 565°C, and R&D is being pursued to achieve higher steam temperatures.

TABLE 2. MATERIALS SELECTED FOR VARIOUS COMPONENTS OF SODIUM-COOLED REACTORS

Country	Reactor	Fuel cladding	Vessel	Primary/secondary piping	IHX	Steam generator	
						Evaporator	Superheater
USA	EBR-II	316	304	304/304	304	Fe-2¼Cr-1Mo	Fe-2¼Cr-1Mo
	FFTF	316SS, HT-9	304	316/316	316	[a]	
UK	DFR	Nb	18-18-1	18-18-1/18-18-1	316	321	321
	PFR	M316SS, PE 16	321	321/321	316	Fe-2¼Cr-1Mo	316H/9Cr-1Mo
Russian Federation	BOR-60		304	304	304	Fe-2¼Cr-1Mo	Fe-2¼Cr-1Mo
	BN-350	Fe-16Cr-15Ni-Mo-Ti-Si	304	304	304	Fe-2¼Cr-1Mo	Fe-2¼Cr-1Mo
	BN-600	Fe-15Cr-15Ni-Mo-Ti-Si	304	304/304	304	Fe-2¼Cr-1Mo	304
France	Rapsodie	316SS	316	316/316	316	[a]	
	Phenix	316SS, 15-15 Ti	316	316/321	316	Fe-2¼Cr-1Mo	321
	Super Phenix	Fe-15Cr-15Ni-Mo-Ti-Si	316	316/316	316L(N)	Alloy 800	
Japan	Joyo	316SS	304	304/Fe-2¼Cr-1Mo	304	[a]	[a]
	Monju	Mod 316SS	304	304/304	304	Fe-2¼Cr-1Mo	304
	JSFR[b]	ODS Ferritic	316FR	9Cr or 12Cr steel		12 Cr steel	
India	PFBR[c]	Fe-15Cr-15Ni-Mo-Ti	316	316L(N)/316L(N)	316	Modified 9Cr-1Mo	

[a] Sodium to air heat exchanger.
[b] In design.
[c] Under construction.

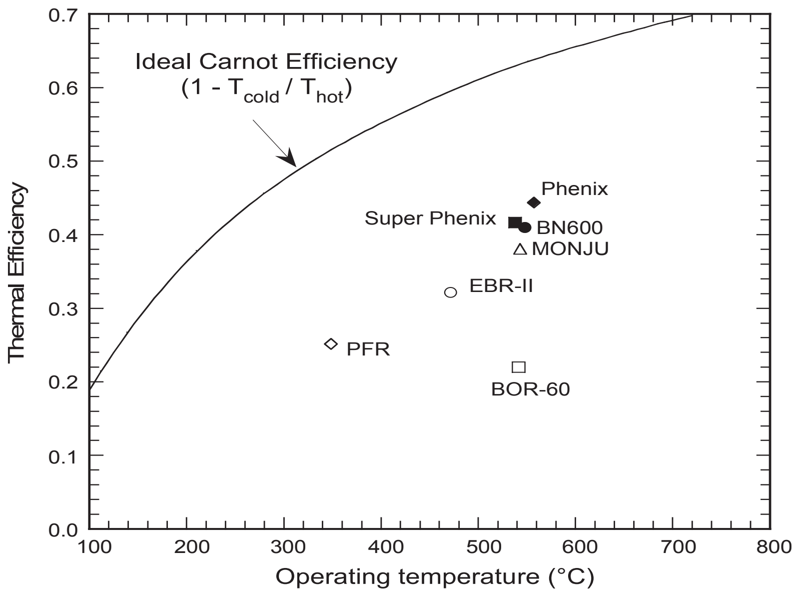


FIG. 2. The ideal thermal efficiency of steam cycle and efficiencies of several SFRs.

3. MATERIALS PERFORMANCE IN SFRS

3.1. Cladding and duct materials

Table 3 gives a listing of the alloys that have been used and those that are being proposed for future use for fuel cladding, ducts, wire wrapper, etc. Historically, 20% cold worked Type 316 stainless steel has been used for fuel cladding. However, owing to significant swelling due to irradiation, its use has been limited to approximately 10% burnup. Subsequently, additional materials such as 15Cr-15Ni-Ti and its Si-modified version, ferritic-martensitic steels such as HT-9, Grade 91 and 92 steels, and oxide dispersion strengthened (ODS) ferritic steels have been or are being tested for their swelling resistance. Figure 3 shows a plot of diametral deformation as a function of dose for 316, Ti-modified 316, and 15Cr-15Ni-Ti, Si-modified 15Cr-15Ni-Ti alloys [2]. Figure 4 shows a comparison of the irradiation effect on the austenitic alloys and ferritic-martensitic steels, including ODS alloys [3]. It is evident that the ferritic steels exhibit superior radiation tolerance; however, additional effort is needed to establish the saturation effect of dose, if any, on deformation damage.

TABLE 3. REQUIREMENTS FOR MATERIALS FOR IN-CORE COMPONENTS IN SFRS

Structure and component	Material	Environment	Degradation process or mechanism	Factors controlling occurrence	Fabrication capability	Knowledge/database
Fuel cladding	Type 316 SS, 15Cr-15Ni-Ti, Si-modified 15Cr-15Ti-Ti, HT-9, Grade 91 and 92 steels, ODS ferritics	Primary Na Argon gas Fission products Fuel constituents	Hardening, swelling, Irradiation creep, Ductility loss, Helium embrittlement Weld integrity	Neutron fluence Service temperature Service life Na compatibility Fuel side compatibility	Good for austenitics Fair to insufficient for ferritics	Adequate for low burnup
Ducts, wire wrapper	Type 316 SS	Primary Na and argon gas	Hardening, swelling Irradiation creep Ductility loss Helium embrittlement Weld integrity	Neutron fluence Service temperature Service life Na compatibility	Good	Adequate for low burnup

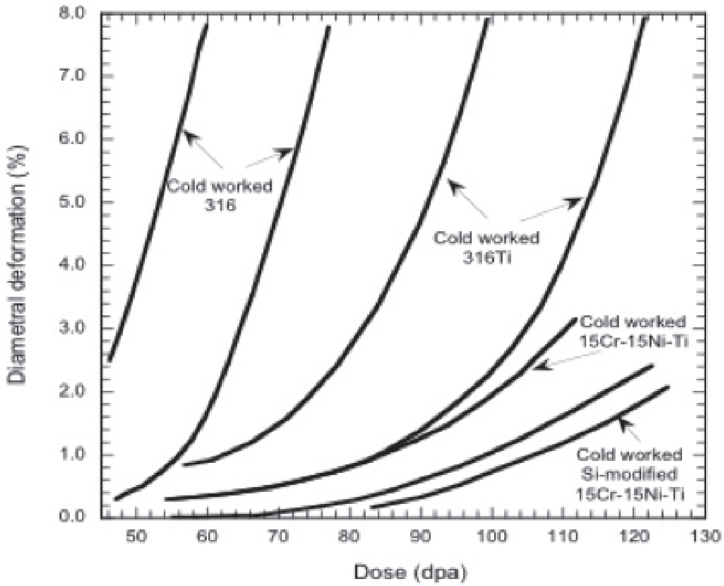


FIG. 3. Variation with dose of the maximum diametral deformation of fuel pins irradiated in Phenix for various cladding materials [2].

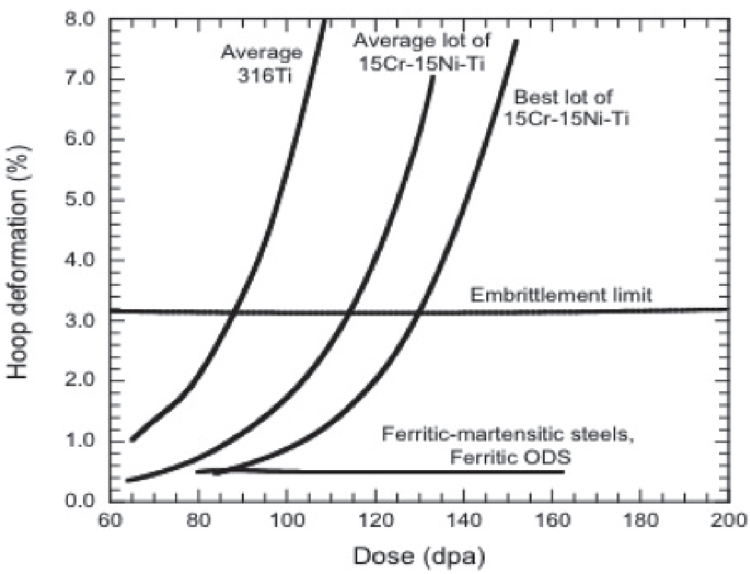


FIG. 4. Maximum hoop deformation of different grades of austenitic and ferritic-martensitic steels, including ODS steels, irradiated in Phenix [3].

Additional issues concerning the factors that control the selection of materials for cladding and core internals are sodium compatibility and fuel side compatibility including the influence of fuel constituents and fisson products. The structural integrity and irradiation performance of weld joints, especially of ferritic-martensitic steels and ODS alloys need significant additional research effort. Figure 5 shows the synergistic effect of irradiation and sodium on the creep rupture properties of Type 316 stainless steel [4]. The Larson-Miller plots of in-air, in-sodium and in-reactor data shown in Fig. 5 indicate that creep rupture strength is not affected by sodium exposure only when the Larson-Miller-Parameter (LMP) <16.5; significant effects of sodium occur above this value. Combined effects of sodium and irradiation cause a significant reduction in rupture strength throughout the LMP range.

3.2. Materials for out-of-core components

Economic competitiveness is a key element in the development of materials for out-of-core components. Advanced materials allow compact and simple design of sodium cooling systems and reactor structure, and have the

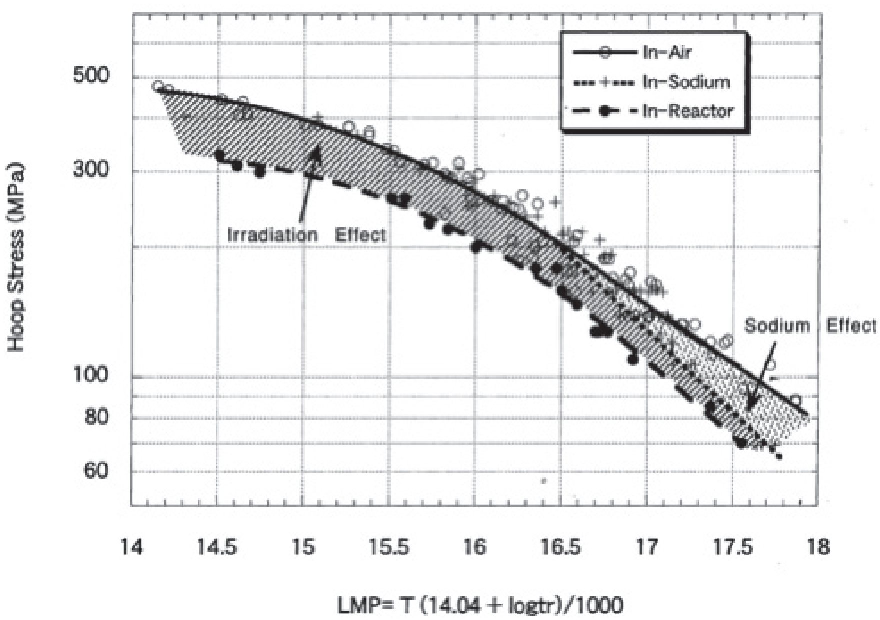


FIG. 5. Effects of sodium and irradiation on creep rupture properties of 316SS [4].

potential to reduce the construction and operational costs of SFRs. It is evident from Table 2 that austenitic stainless steels and low alloy ferritic steels have been used exclusively in the past reactors for reactor vessel, primary/secondary piping, IHX and steam generators. Nuclear structural component construction in the US complies with Section III of the ASME Boiler and Pressure Vessel Code. As the SFR has an outlet temperature of >500°C and a 60-year design life, significantly different from the current light water reactors, the design of elevated temperature components must comply with ASME Section III Subsection NH, and also must consider time dependent effects on mechanical properties such as creep, creep-fatigue and creep ratcheting. Subsection NH provides the high temperature design rules for components in nuclear service. The rules were developed in support of the US LMFBR programme in the 1960s and 1970s. Subsection NH has a rather limited choice of materials, with only five materials qualified that include Types 304 and 316 austenitic stainless steels, 2.25Cr-1Mo steel, modified 9Cr-1Mo, and Alloy 800H. Therefore, in addition to developing adequate data for new/advanced materials such as Grade 92 steel and Alloy NF709, high temperature design rules must also be developed to account for high temperature damages, such as creep rupture, excessive creep deformation, creep buckling, cyclic creep ratcheting and creep fatigue, and consider environmental effects. A detailed in-depth assessment of materials qualification and licensing needs was conducted earlier [5]. Several of the technical issues and data gaps for existing materials were addressed in a later report [6]. However, issues such as data extrapolation to 60 years (525 000 hours) design life from the current 300 000 hours life, creep-fatigue behaviour of materials with long hold times, weld qualification and the basis for weld reduction factors, etc., need resolution. These issues are currently being addressed in ASME Section III, Division 5, directed towards high temperature reactors.

3.2.1. Reactor system

Austenitic stainless steels have been widely used for fast reactor applications, including reactor core, reactor vessel, grid plate, sodium piping, primary heat exchangers and sodium pumps. Austenitic stainless steels have excellent high temperature mechanical properties and corrosion resistance, good fabrication capability and weldability. Compared to ferritic steels, they are more expensive and have higher thermal expansion coefficient and lower thermal conductivity.

The NRC licensing review of the Clinch River Breeder Reactor (CRBR) and the Power Reactor Innovative Small Module (PRISM) project identified a number of technical issues even for the materials that have been used in past reactors (see Table 2). These include embrittlement issues pertinent to a 60-year lifetime, sensitization and stress corrosion cracking at elevated temperatures, effects of secondary phases, hot cracking and creep-fatigue fracture, erosion–corrosion and property degradation in a sodium environment, weldment safety evaluation, etc., and many of these issues are currently being addressed. Table 4 gives a listing of materials for various reactor structural components, along with anticipated service environments, potential damage mechanisms, and factors that may control the damage occurrence. It is anticipated that the components in current and future sodium reactors will predominantly utilize the austenitic stainless steels.

3.2.2. Primary and secondary heat transport systems

The primary heat transport system is used to transfer the heat from the core to the secondary sodium transport system via an intermediate heat exchanger. Table 5 gives a listing of materials for various structural components in the primary heat transport system, along with anticipated service environments, potential damage mechanisms, and factors that may control the damage occurrence. Predominantly austenitic stainless steels have been used in all the past reactors. In the loop type JSFR, it is planned to use high Cr ferritic (9 or 12Cr steels, with high strength and low thermal expansion coefficient) for the piping and heat exchanger. In addition, variants of Type 316 stainless steels such as 316FR and 316L(N) are being planned for use in the JSFR (Japan) and PFBR (India), respectively. Figure 6 shows a comparison of the creep data for the conventional 316, 316L(N) and 316FR in the temperature range 500–600°C [7].

3.2.3. Material performance in sodium environments

Transport/deposition of non-metallic elements such as carbon and nitrogen in sodium circuits can have a significant effect on mechanical properties of austenitic stainless steels as well as ferritic and ferritic-martensitic steels. In a monometallic sodium system, austenitic stainless steels decarburize in the high temperature region and carburize in the low temperature region; in bimetallic sodium circuits constructed of austenitic and ferritic steels, the ferritic steel located in the low temperature region tends to decarburize and the austenitic stainless steel located in the high temperature region tends to carburize.

TABLE 4. REQUIREMENTS FOR REACTOR SYSTEM STRUCTURAL COMPONENTS

Structure and component	Material	Environment	Degradation process or mechanism	Factors controlling occurrence	Fabrication capability	Knowledge/database
Vessel	Type 316 SS	Primary Na and argon gas	Thermal ageing Thermal creep Weld integrity	Neutron fluence Service temperature Service life	Good	Adequate 60-year service?
Vessel enclosure	Type 316 SS	Primary Na and argon gas	Thermal ageing Thermal creep Weld integrity	Neutron fluence Service temperature Service life	Good	Sufficient 60-year service?
Guard vessel	Type 316 SS Fe-9Cr-Mo Steel	Argon gas and leaking Na	Corrosion	Temperature Service life Na purity	Good	Sufficient 60-yr service?
Core support structure	Types 304 and 316 SS	Primary Na	Irradiation Thermal ageing Crevice corrosion	Temperature Neutron fluence Service life	Good	Sufficient 60-yr service?

TABLE 5. REQUIREMENTS FOR PRIMARY AND SECONDARY HEAT TRANSPORT SYSTEM COMPONENTS

Structure and component	Material	Environment	Degradation process or mechanism	Factors controlling occurrence	Fabrication capability	Knowledge/database
Primary and secondary piping	Type 316, High Cr steels	Primary Na Secondary Na	Interstitial element transfer Thermal ageing Fatigue resistance Radioactive mass transport (primary piping only)	T and Delta T Na purity Mechanical load Na flow velocity	Good	Fairly good Needs system assessment regarding dynamic carbon level
Mechanical pump (impeller, diffuser)	Type 316	Primary or secondary Na	Corrosion Fatigue resistance	Flow velocity Vibration Applied load Na purity Temperature	Good	Fairly good
EM pump	To be determined	Primary or secondary Na	Corrosion Fatigue resistance Electrical compatibility	Flow velocity Applied load Na purity Temperature	Unknown	Inadequate
IHX shell	Type 304 or 316	Primary Na inside	Thermal ageing Fatigue and creep-fatigue Interstitial element transfer	Na purity Service life Temperature Mechanical load	Good	Adequate

TABLE 5. REQUIREMENTS FOR PRIMARY AND SECONDARY HEAT TRANSPORT SYSTEM COMPONENTS (cont.)

Structure and component	Material	Environment	Degradation process or mechanism	Factors controlling occurrence	Fabrication capability	Knowledge/database
IHX tubes	Type 304H, High Cr steels	Secondary Na inside and primary Na outside	Thermal creep Fatigue and creep-fatigue Interstitial element transfer	Na purity Service life Temperature Mechanical load	Good	Adequate for 304H Insufficient for high Cr steels

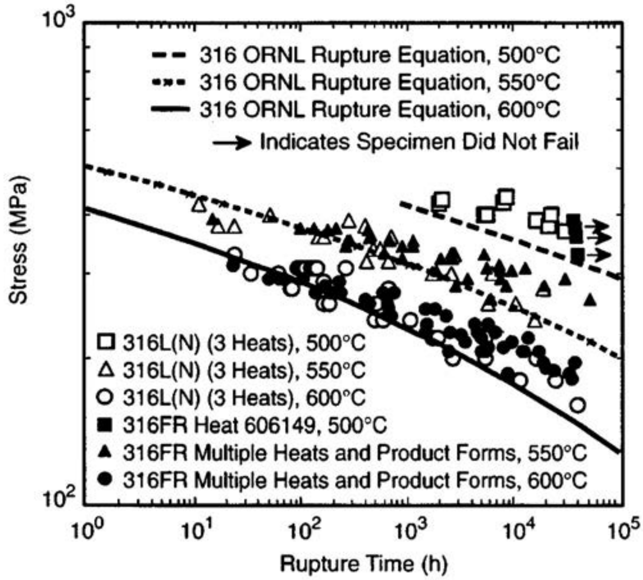


FIG. 6. Stress-rupture data for 316FR, 316LN and 316 stainless steels [7].

The carbon activity in sodium and the extent of carburization/decarburization depend on the temperature, temperature distribution around the sodium circuit, material composition, and thermomechanical treatments as well as external carbon sources and sinks. Figure 7 gives an example of carburization/decarburization in 304 SS and 316 SS in EBR-II and FFTF environments [8]. The carbon concentration-penetration-temperature diagram shown in Fig. 8 represents the influence of time, temperature, carbon concentration in sodium, and thermomechanical treatment on the carburization/decarburization behaviour of Types 304 and 316 SS [8–10]. The region of maximum carburization is indicated by line A, and the penetration depth for the transition between single-phase austenite and a two-phase austenite plus carbide mixture is represented by line B. The variation of the carbon concentration at the surface of the steel with temperature is shown in the plane corresponding to zero penetration in the material.

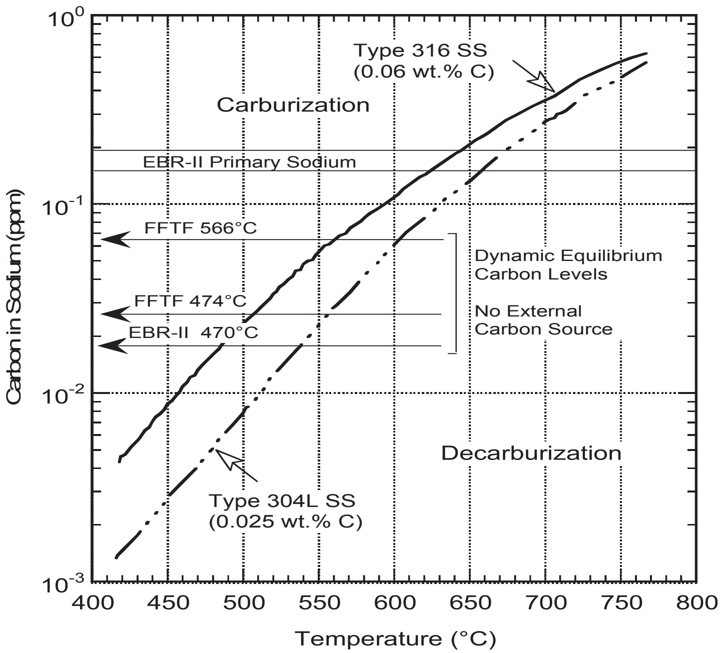


FIG. 7. Carburization/decarburization in austenitic steels on temperature and C in sodium [8].

Sodium appears to have a more significant effect on the creep properties of 304 SS than on 316 SS. Degradation of creep properties of Type 304 SS in sodium has been reported in several studies [11–14]. In a carburizing sodium environment, i.e. when C activity in sodium is greater than that in steel, reduction in rupture life occurs due to reduced ductility in the tertiary creep regime. Tertiary creep embrittlement is more severe at higher test temperatures. Minimum creep rate and time-to-onset of tertiary creep are not affected by a carburizing sodium environment. In a decarburizing sodium environment, enhanced creep rate and earlier onset of tertiary creep were observed in addition to the shortened rupture time. Cracking along the affected grain boundaries leads to reduction in load bearing cross-section that, in turn, shortens the rupture time for a given applied stress.

Sodium exposure has nearly no effect on the creep rupture properties of Type 316 SS, as shown in Fig. 9 [15–17]. Creep rupture properties of Type 316 SS showed little change after 10 000 hours exposure to sodium at 593°C. This is probably due to the presence of fine molybdenum carbides in 316 steel. The creep-rupture strength of the specimens tested at ANL was lower than the ASME Code minimum owing to a low N content of the steel. The sodium

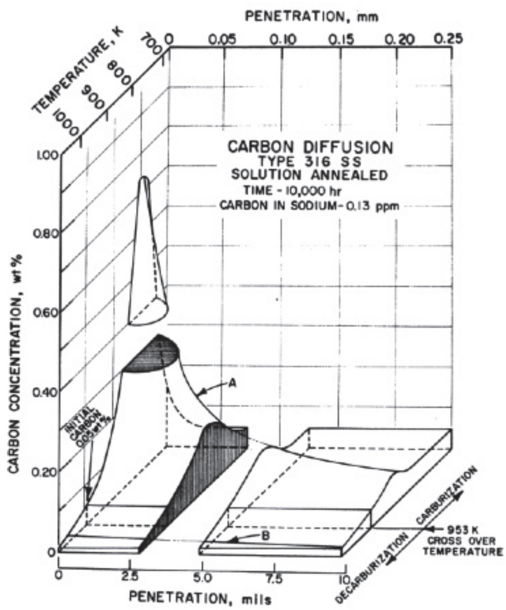


FIG. 8. Effect of temperature on the carburization/decarburization of Type 316 SS [9].

effect was observed in sensitized 316L(N), exhibiting two stages of secondary creep during long term creep tests (>6000 h) [18].

The effect of sodium exposure on the fatigue properties of austenitic stainless steels has been evaluated by fatigue testing in flowing sodium and testing specimens pre-exposed to sodium environments. A beneficial effect of sodium on fatigue life was observed in austenitic stainless steels when tested under continuous fatigue loading. The fatigue life of Type 316 SS in sodium is significantly longer than in air, as shown in Fig. 10 [19–23]. For the sodium exposure conditions used in these studies, the specimens developed a carbon concentration profile that varied from ≈0.4wt% at the surface to the initial concentration in the steels at a depth of 0.01–0.02 cm. The results indicate that moderate carburization of the steel in sodium has negligible effect on the fatigue life and that the fatigue lives of the sodium exposed material are comparable to those of the annealed or thermally aged material. Surface oxidation, when tested in air, may facilitate early crack initiation, leading to shortened fatigue life. The absence of surface oxidation in sodium increases the fatigue life considerably, when tested in fully reversed fatigue loading.

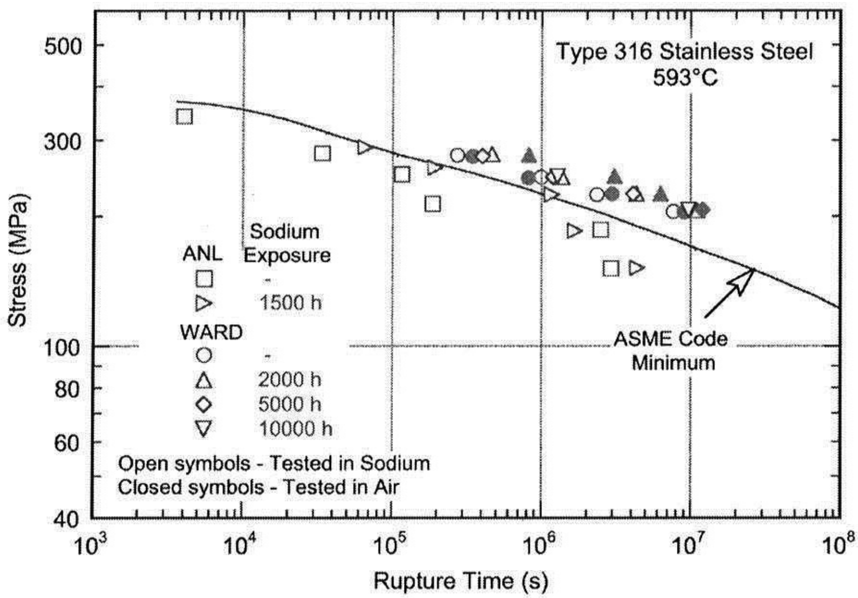


FIG. 9. Effect of sodium exposure on creep rupture properties for Type 316 SS [14–17].

3.2.4. Steam generator components in SFRs

Table 6 gives a listing of key components in a steam generator and pertinent details on the service environment and damage mechanisms. Historically, Fe-2¼Cr-1Mo steel for evaporators and austenitic stainless steels/Alloy 800 for superheaters were selected (see Table 2). A major concern with Fe-2¼Cr-1Mo steel for this application is the susceptibility of the steel to decarburization when exposed to high temperature flowing sodium [24–26]. The C loss from the material leads to a significant reduction in elevated temperature mechanical strength [27, 28]. Furthermore, in sodium systems that contain austenitic and ferritic components, decarburization of the ferritic material leads to carburization of the austenitic SSs [29–31].

Information on the kinetics of decarburization of Fe-2¼Cr-1Mo steel in a sodium environment has been used to calculate the average C concentrations of a typical steam generator tube (initial C content of 0.12wt% and 3.68 mm wall thickness) as a function of service temperature after a 30-year period. The results are presented in Fig. 11 along with the approximate temperature ranges for the steam generators in the EBR-II, Phenix and Super Phenix plants. Figure 12 is a plot of C loss from the steel after 30 years operation as a function of tube wall

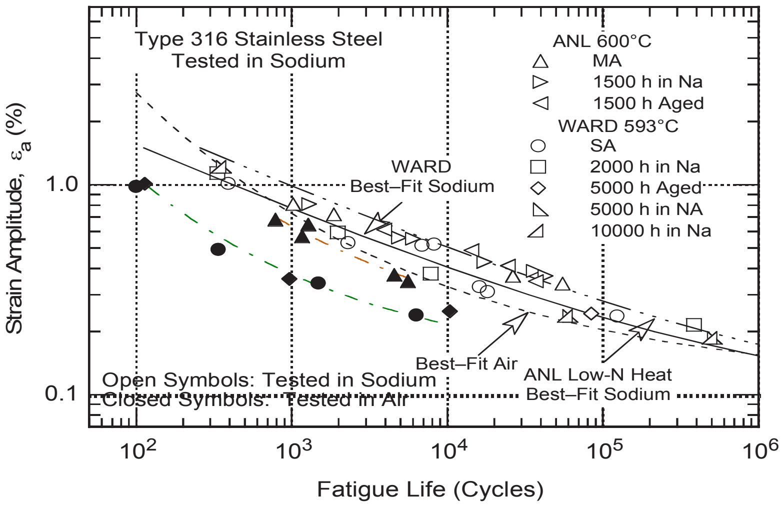


FIG. 10. Fatigue strain–life relation for Type 316 SS in air and in sodium environments [19–23].

thickness and temperature. The amount of C loss is greater for tubes with thinner wall sections, and increases with temperature.

The high Cr ferritic steels provide a greater resistance to C transfer [32] and possess adequate elevated temperature mechanical properties. The equilibrium relationships indicate that, at a C concentration in sodium of 0.05 ppm at 550°C (which corresponds to a C activity of 0.017 in sodium), the Fe-2.25Cr-1Mo and Fe-9Cr-Mo steels would contain ≈0.027 and 0.092wt% C, respectively. The microstructure as well as the composition of the steel plays an important role in the carburization/decarburization behaviour. For example, stabilizing elements such as Nb and V in Fe-9Cr-Mo steel increase the equilibrium C concentration in the steel for a given C activity of the sodium.

TABLE 6. REQUIREMENTS FOR POWER CONVERSION SYSTEM, STEAM RANKINE CYCLE COMPONENTS

Structure and component	Material	Environment	Degradation process or mechanism	Factors controlling occurrence	Fabrication capability	Knowledge/Database
SG shell	Fe-2 1/4Cr-1Mo	Secondary Na	Interstitial transfer Thermal ageing Sodium–water reaction Thermal creep Fatigue and creep-fatigue	Na purity T, Delta T Steam leak Applied load Ageing time	Good	Fairly good
SG tubing	Fe-2 1/4Cr-1Mo	Water or steam inside and secondary Na outside	Na corrosion C transfer Thermal ageing Caustic effect Thermal creep Fatigue and creep-fatigue	Na purity T & Delta T Steam leak Applied load Transients Ageing time	Good	Fairly good
Hot leg steam piping	Fe-2 1/4Cr-1Mo	Steam	Flow assisted corrosion Fatigue and creep-fatigue Thermal ageing	Flow velocity Steam T & P Service time Ageing time Water chemistry	Good	Fairly good
Cold leg steam piping	Carbon steel	Treated water	Flow assisted corrosion General corrosion Fatigue	Flow velocity Steam pressure Temperature Service time Ageing time Water chemistry	Good	Fairly good

TABLE 6. REQUIREMENTS FOR POWER CONVERSION SYSTEM, STEAM RANKINE CYCLE COMPONENTS (cont.)

Structure and component	Material	Environment	Degradation process or mechanism	Factors controlling occurrence	Fabrication capability	Knowledge/Database
Steam turbine	Ferritic steel (intermediate chromium)	Steam	Steam oxidation Scale exfoliation Creep and fatigue Flow induced corrosion	Temperature Service time Applied load Steam velocity	Good	Fairly good

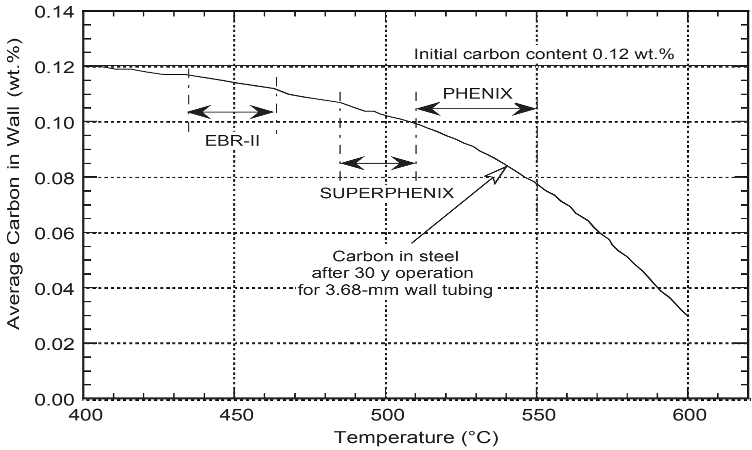


FIG. 11. Calculated average C concentration in 2.25Cr-1Mo steel tubing as a function of temperature after 30 years of exposure to sodium.

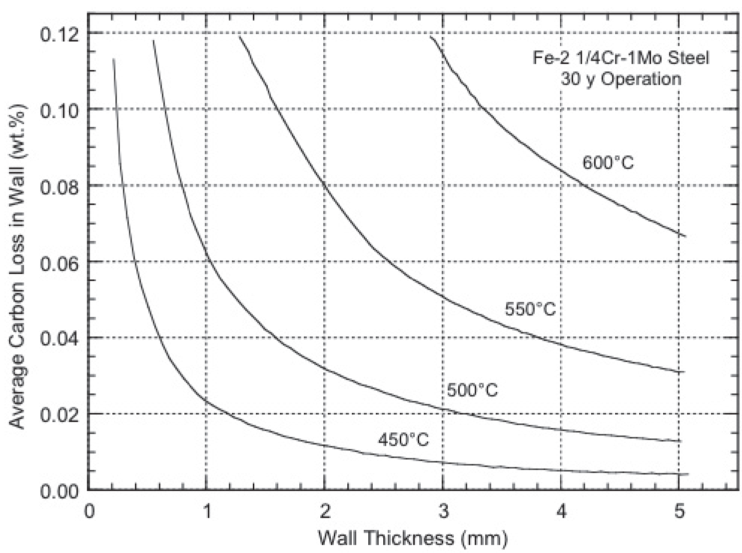


FIG. 12. Calculated average C loss as a function of tube wall thickness and temperature for Fe-2¼Cr-1Mo steel after 30 years of exposure to sodium.

Figure 13 shows little difference in creep strength between as received and sodium exposed (with simultaneous thermal ageing) specimens of mod.9Cr-1Mo [15, 33]. No effect of pre-sodium exposure (associated carbon gain/loss) was observed after 5000 hours exposure to sodium at 500–550°C. The creep-fatigue life (see Fig. 14) of mod.9Cr-1Mo steel under tensile hold was significantly shorter than fatigue life under continuous cycling, both in the sodium environment and in air, and no beneficial effect of sodium exposure on fatigue life was observed, while a compressive hold in sodium was less damaging [33]. Steels of 9Cr-1Mo and/or 12Cr type have been selected for steam generators in the PFBR (under construction) and in the JSFR (in conceptual design).

4. SUMMARY

The present paper identifies the materials requirements for various in-core and out-of core components for SFRs and assesses the materials that have been used in the past experimental and demonstration reactors. The degradation processes and the factors that control degradation in the service environment are key factors in the selection of materials and their long term performance. The primary goal of materials for in-core applications is radiation tolerance and life extension of fuel cladding and surrounding components. To achieve this goal, materials such as ferritic-martensitic steels, including ODS alloys, are under development. For out-of-core components such as primary/secondary piping,

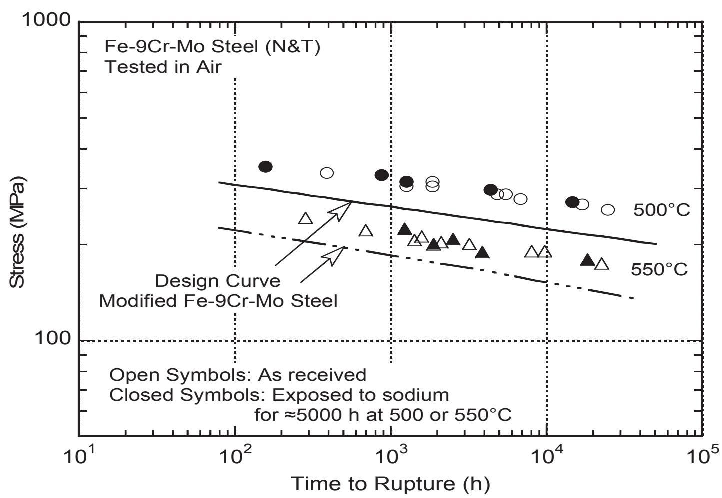


FIG. 13. Creep rupture data for 9Cr-Mo steels in air and in sodium [34].

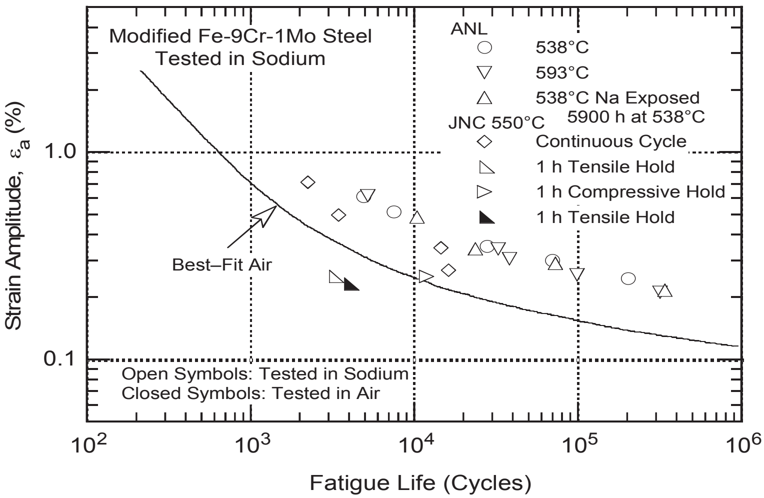


FIG. 14. Effects of sodium on fatigue and creep-fatigue behaviour of mod. 9Cr-1Mo steel [34].

IHX, etc., austenitic stainless steels of several types have been used in the past reactors and probably will continue to be used in future reactors along with the ASME Code qualified material such as mod.9Cr-1Mo steel. Advanced materials such as Grade 92 steel and NF709, owing to their relatively higher strength, can play a significant role in reducing the piping wall thickness thereby reducing the commodity requirements and decreasing the capital cost of the plant. The materials used in the past for steam generators have been low alloy steel and austenitic stainless steel. If an increase in steam temperature is desired along with a design life of 60 years, the sodium compatibility and thermal ageing of the current materials need reassessment and advanced materials with higher strength and higher temperature capability are warranted. In addition, if a supercritical CO_2 power conversion system involving the Brayton cycle is considered, significant effort is needed to address the compatibility of materials in the high pressure CO_2 environment and to develop a methodology for the design of sodium–CO_2 heat exchanger in compliance with the ASME Code.

ACKNOWLEDGEMENTS

This work was supported by the US Department of Energy, Office of Nuclear Energy, Advanced Reactor Concepts (ARC) and Small Modular Reactor (SMR) Programs.

REFERENCES

[1] WIKIPEDIA, Fast Breeder Reactor, http://en.wikipedia.org/wiki/Breeder_reactor

[2] SERAN, J.L., LEVY, V., DUBUISSON, P., GILBON, D., MAILLARD, A., FISSOLO, A., TOURON, H., CAUVIN, R., CHALONY, A., LE BOULBIN, E., Proc. 15th Int. Symp. Effects of Radiation Materials: ASTM STP 1125 (STOLLER, R.E., KUMAR, A.S., GELLES, D.S., Eds), American Society for Testing and Materials, Philadelphia (1992) 1209.

[3] YVON, P., CARRE, F., J. Nucl. Mater. **385** (2009) 217.

[4] UKAI, S., MIZUTA, S., KAITO T., OKADA, H., J. Nucl. Mater. **278** (2000) 320.

[5] NATESAN, K.M., LI, S., MAJUMDAR, R., NANSTAD, K., SHAM., T.-L., Argonne National Laboratory, Rep. ANL-AFCI-244, Argonne Natl. Lab., IL (2008).

[6] NATESAN, K.M., LI, S. MAJUMDAR, R., NANSTAD, K., SHAM, T.-L., Argonne National Laboratory, Rep. ANL-AFCI-285, Argonne Natl Lab., IL (2009).

[7] BRINKMAN, C.R., J. Press. Vessel Technol. **123** (2001) 75–80.

[8] SNYDER, R.B., NATESAN, K., KASSNER, T.F., Proc. Intl. Conf. on Liquid Metal Technology in Energy Production, CONF-760503-P2, Champion, PA, 1976 (1976) 826.

[9] SNYDER, R.B., NATESAN, K., KASSNER, T.F., A Generalized Method of Computing C-Diffusion Profiles in Austenitic Stainless Steels Exposed to a Sodium Environment, Argonne National Laboratory, Rep. ANL-8015, Argonne Natl Lab., IL (1973).

[10] SNYDER, R.B., NATESAN, K., KASSNER, T.F., J. Nucl. Mater. **50** (1974) 259.

[11] WADA, Y., YOSHIDA, E., AOKI, M., KATO, S., ITO, T., IAEA Specialists Mtg IWGFR-84, KFK 4935 (BORGSTEDT, H.U., KARSLRUHE, F.R.G., Eds), Kernforschungszentrum Karlsruhe, Germany (1991) 17.

[12] BORGSTEDT, H.U., FREES, G., HUTHMANN, H., IAEA Specialists Mtg IWGFR-84, KFK 4935 (BORGSTEDT, H.U., KARSLRUHE, F.R.G., Eds), Kernforschungszentrum Karlsruhe, Germany (1991) 86.

[13] NATESAN, K., CHOPRA, O.K., KASSNER, T.F., Proc. Int. Conf. Liquid Metal Technology in Energy Production, CONF- 760503-Pl, Champion, PA, 1976 (1976) 338.

[14] HUTHMANN, H., MENKEN, G., BORGSTEDT, H.U., TAS, H., Proc. 2nd Int. Conf. on Liquid Metal Technology in Energy Production, CONF-800401, Richland, WA, 1980 (1980) 19–33.

[15] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, Boiler and Pressure Vessel Code - 2007 Edition, New York, NY (2007).

[16] NATESAN, K., CHOPRA, O.K., ZEMAN, G.J., SMITH, D.L., KASSNER, T.F., “Effect of Sodium Environment on the Creep-Rupture and Low-Cycle Fatigue Behavior of Astenitic Stainless Steels”, Proc. IAEA Specialists Mtg on Properties of Primary Circuit Structural Materials including Environmental Effects, CONF-771052, Bergisch Gladbach, Germany, 1977, IAEA, Vienna (1977) 168.

[17] FLAGELLA, P.N., DENNE, J.A., LEISURE, R.A., Proc. 2nd Int. Conf. Liquid Metal Technology in Energy Production, CONF-800401-P2, Richland, WA (1980) 19–50.

[18] HUTHMANN, H., BORGSTEDT, H.U., DEBERGH, PH., HORTON, C.A.P., WODD D.S., IAEA Specialists Mtg IWGFR-84, KFK 4935 (BORGSTEDT, H.U., KARSLRUHE, F.R.G., Eds), Kernforschungszentrum Karlsruhe, Germany (1991) 102.
[19] SMITH, D.L, ZEMAN, G.J., NATESAN, K., KASSNER, T.F., Proc. Int. Conf. Liquid Metal Technology in Energy Production, CONF-760503-Pl, Champion, PA, 1976 (1976) 359.
[20] SMITH, D.L., NATESAN, K., KASSNER, T.F., ZEMAN, G.J., American Society of Mechanical Engineers Symp. Structural Materials for Service at Elevated Temperatures in Nuclear Power Generation, MPC-1, Houston, TX (1975) 290.
[21] ZEMAN, G.J., SMITH, D.L., J. Nucl. Technol. **42** (1979) 82.
[22] FLAGELLA, P.N., KAHRS, J.R., Proc. Int. Conf. on Liquid Metal Technology in Energy Production, CONF-760503-Pl, Champion, PA, 1976 (1976) 353.
[23] NATESAN, K., CHOPRA, O.K., KASSNER, T.F., Proc. 2nd Int. Conf. on Liquid Metal Technology in Energy Production, CONF-800401, Richland, WA, 1980 (1980) 19–41.
[24] NATESAN, K., CHOPRA, O.K., KASSNER, T.F., J. Nucl. Technol. **28** (1976) 441.
[25] KRANKOTA, J.L. ARMIJO, J.S., J. Nucl. Technol. **24** (1974) 225.
[26] MATSUMOTO, K., OHTA, Y., KATAOKA, T., YAGI, S., SUZUKE, K., YUKITOSHI, T., MOROISHI, T., YOSHIKAWA K., SHIDA, Y., J. Nucl. Technol. **28** (1976) 452.
[27] KRANKOTA, J.L., ARMIJO, J.S., Proc. Sym. Low C and Stabilized 2¼% Cr-1% Mo Steels, American Society for Metals, Metals Park, Ohio (1973) 57.
[28] SPONSELLER, B.L., N. SEMCHYSHEN, P.J. GROBNER, Proc. Sym. Low C and Stabilized 2¼%Cr-1%Mo Steels, American Society for Metals, Metals Park, OH (1973) 73.
[29] ATSUMO, H., YUHARA, S., MARUYAMA, A., KANOH, S., AOKI, N., MOCHIZUKI, K., Proc. Int. Conf. on Liquid Metal Technology in Energy Production, CONF-760503-P2, Champion, PA, 1976 (1976) 849.
[30] BESSON, M., BAQUE, P., CHAMPEIX, L., DONATI, J.R., OBERLIN, C., SAINT-PAUL, P., Proc. Int. Conf. on Liquid Metal Technology in Energy Production, CONF-760503-P2, Champion, PA, 1976 (1976) 834.
[31] KRANKOTA, J. L. CHALLENGER, K.O., Proc. Int. Conf. on Liquid Metal Technology in Energy Production, CONF-760503-P2, Champion, PA, 1976 (1976) 730.
[32] MENKEN, G., GROSSER, E.D., TE HESSEN, E., Proc. Int. Conf. on Ferritic Steels for Fast Reactor Steam Generators, BNES, London, 1978 (1978) 264.
[33] ASAYAMA, T., ABE, Y., MIYAJI, N., KOI, M., FURUKAWA, T., YOSHIDA, E., J. Press. Vessel Technol. **123** (2001) 49.

ACHIEVEMENT AND NEW CHALLENGES FOR HIGH PERFORMANCE MATERIALS IN EUROPE

C. FAZIO
Karlsruhe Institute of Technology,
Karlsruhe, Germany

P. DUBUISSON
Commissariat à l'énergie atomique et aux énergies alternatives,
Saclay, France

Abstract

The European approach in the framework of nuclear energy advancement is described, in the light of such prominent objectives as safety, sustainability and transmutation of nuclear waste. The highlights are given on innovation and safety related design aspects of relevant fast neutron reactor component and systems. The materials requirements and related new challenges on materials for selected systems as decay heat removal systems, energy conversion systems, core components, etc., will be discussed and related European R&D programmes presented.

1. INTRODUCTION

Materials development is a key scientific and technological area for the advancement of nuclear energy. In Europe, the technology platform on sustainable nuclear energy (SNETP) [1] has identified three research and innovation pillars for nuclear energy. These three pillars are: (1) the safe operation of presently operating light water reactors, (2) the development of new applications of nuclear power such as industrial heat applications and (3) the development of a new generation of more sustainable reactor technologies.

The third pillar is of relevance for this manuscript since it addresses the development of fast neutron systems with closed fuel cycle. In this framework, the SNETP has launched an industrial initiative for sustainable energy (ESNII) [2]. The ESNII contemplates three fast reactor (FR) technologies, i.e. the liquid sodium technology as the reference one, since more experience is available in this field, and the liquid lead/lead–bismuth and gas technologies as alternative options. For each of these technologies a specific reactor design was proposed and each one has specific materials challenges. These challenges are addressed in the context of national programmes and EURATOM supported projects such as Generation IV and Transmutation Materials (GETMAT) [3, 4] , Materials Testing and Rules (MATTER) [5], etc., and by the Joint Program Nuclear Materials [6] initiated by the European Energy Research Alliance (EERA) [7]. This manuscript is structured so as to give, first, a brief description of the ESNII systems and related key materials issues, after which relevant experimental programmes are presented and discussed. The manuscript will be concluded with a summary and the outlook.

2. ESNII FACILITIES

As shown in Fig. 1, in the framework of ESNII four different reactor systems are under development:

— ASTRID (Advanced Sodium Technological Reactor for Industrial Demonstration) developed by the CEA in close collaboration with AREVA and EdF and international partners [8] and considered as the ESNII reference system.
— MYRRHA (Multi-purpose hYbrid Research Reactor for High-tech Applications) developed by SCK-CEN in close collaboration with international partners and considered as a pilot plant for the lead fast reactor (LFR); and ALFRED (Advanced Lead Fast Reactor European Demonstrator), developed in the frame of the European Project LEADER [9] and considered as the demonstrator plant for LFR.
— ALLEGRO (European Gas Fast Reactor (GFR) Demonstrator) developed in the framework of the European Project GoFastR [10] and considered as the demonstrator plant for GFR.

2.1. Materials selection for ASTRID

ASTRID is a pool type sodium cooled fast reactor (SFR) of 1500 MW(th), generating 600 MW(e) and considered as a demonstration system, where innovative design options will be implemented [11]. The design of ASTRID is such as to comply with the Generation IV requirements on safety, sustainability, economic competitiveness and proliferation resistance. Moreover, it is also foreseen as performing irradiation experiments within ASTRID for innovative materials and fuels as well as relevant demonstration tests.

As shown in Fig. 2, ASTRID has the typical components of a pool type SFR, i.e. the core, the diagrid and the redan to separate the cold from the hot

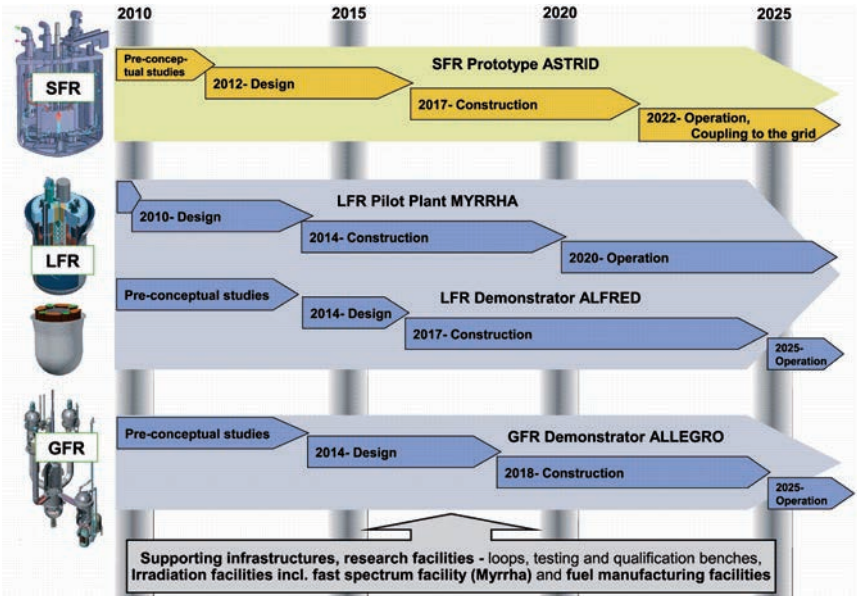


FIG. 1. ESNII roadmap [1, 2].

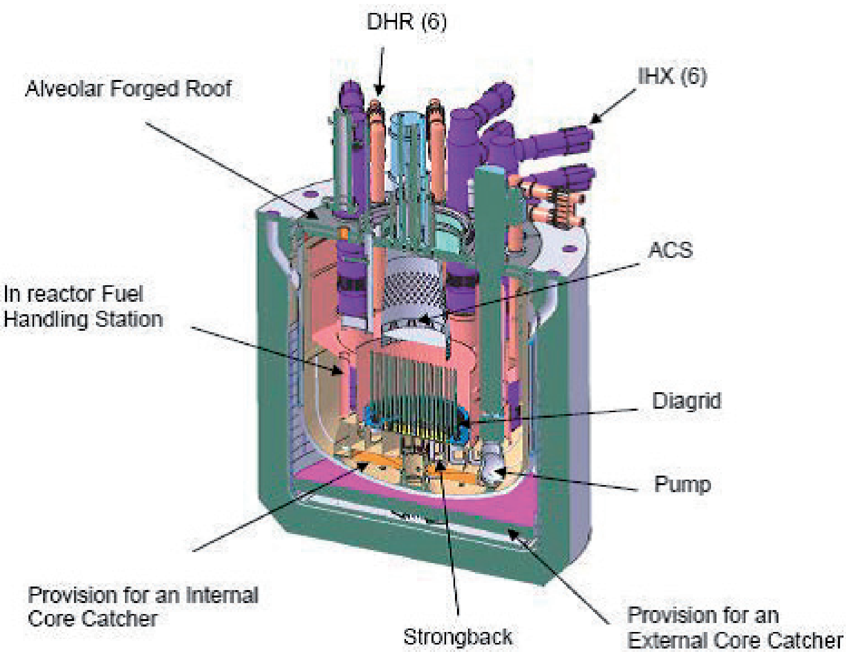


FIG. 2. Pre-conceptual design for ASTRID reactor block.

part, the primary pumps and heat exchangers, etc. The intermediate heat removal system is a sodium cooled one, while for the energy conversion system the options are open between water/steam and gas. The core will be fuelled with $UPuO_2$ fuel annular pellets and the targeted average burnup is higher than 100 GW·d/t.

The material items for the ASTRID reactor as discussed within the MATTER Project and presented at the IAEA TW in Argonne [12, 13] are defined within the French national programme (Table 1), where selected components are described on the basis of the Refs [12, 13].

2.2. Materials selection for MYRRHA and ALFRED

The heavy metal systems considered within Europe are the lead–bismuth eutectic (LBE) cooled system named MYRRHA and the lead (Pb) cooled system named ALFRED. MYRRHA is developed as a multipurpose research facility which can operate in critical and sub-critical modes [14]. For the ESNII purposes, MYRRHA can serve as a test bench for LFR concepts. Moreover, MYRRHA will demonstrate the feasibility of accelerator driven systems (ADS) and the application of LBE technology to nuclear reactors. In Fig. 3(a), the MYRRHA layout is shown and in Fig. 3(b), the layout of ALFRED.

In Ref. [15], the selection of materials and main issues for MYRRHA has been indicated. Moreover, the materials qualification programme for MYRRHA shows commonalities with the corresponding work for the ALFRED system. The structural materials issues to be considered are related to design and safety requirements, where gaps and open questions concerning materials performance under representative conditions (e.g. high temperature, high burnups, and aggressive environment) are identified.

Among the various items listed in Ref. [15] as key issues, the mitigation of corrosion by liquid metal, the degradation of the mechanical properties due to the liquid metal and the irradiation effects should be considered. In addition, similar mechanical effects as listed for the various components in Table 1 might also be identified for MYRRHA and ALFRED. The reference candidate materials for relevant ALFRED components are given in Table 2. It should be considered that for ALFRED, being cooled with Pb, the design temperature at the core inlet is ~400°C and at the core outlet ~480°C; moreover the estimated clad peak temperature is in the order of 550°C. Since MYRRHA is cooled with LBE, the core inlet and outlet temperatures are lower, i.e. ~270°C and ~400°C respectively and the clad peak temperature is in the order of 470°C [16].

TABLE 1. FEATURES OF SOME ASTRID COMPONENTS

Component	Operating conditions	Damage mode to be avoided	Reference materials
Steam generator	525°C (Na); 490°C (steam)	Ageing, deformation, ratcheting, creep-fatigue, buckling, faulting, compatibility with coolants Welding issues	AISI 316LN 800 SPX 9Cr F/M
Pipes and circuits	350–550°C	Creep-fatigue, thermal fatigue, ageing, welding issues	AISI 316LN 9Cr F/M
Intermediate heat exchanger	400–550°C	Creep-fatigue, ageing	AISI 316LN
Primary vessel/Redan	395°C/400–550°C	Deformation, thermal-creep, ratcheting, buckling, fatigue welding, ageing	AISI 316LN
Above core structure	550°C Low irradiation effects, gas production	Creep-fatigue, thermal striping	AISI 316LN
Diagrid	400°C Low irradiation dose		AISI 316LN
Core structures (wrapper/cladding)	400–700°C High irradiation dose	Deformation Swelling, creep, hoop stress Thermal creep	First core: / EM10 / 15-15 Ti AIM1 Long term R&D: advanced martensitic steels, SiC/SiC composite for wrapper and advanced austenitic steels, ODS

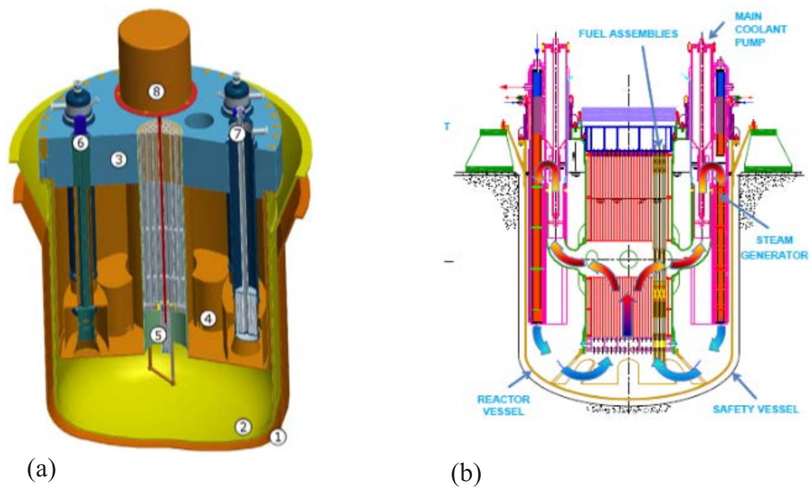


FIG. 3. (a) Layout of MYRRHA [14], (b) layout of ALFRED [9].

TABLE 2. MATERIALS SELECTION FOR THE PB COOLED ALFRED REACTOR [17]

Components	Material
Reactor vessel	AISI 316L
Vessel support	AISI 316L
Safety vessel (cavity liner)	AISI 316L
Reactor cover	AISI 316L
Inner vessel	AISI 316LN
Core lower grid	AISI 316LN
Core upper grid	AISI 316LN
Steam generator	T91
Primary pump: duct and shaft	AISI 316LN

TABLE 2. MATERIALS SELECTION FOR THE PB COOLED ALFRED REACTOR [17] (cont.)

Components	Material
Primary pump: impeller	Needs to be defined. At present MAX phases are investigated
Fuel assembly: cladding	15-15/Ti (coatings are considered for corrosion protection)
Fuel assembly: grids	15-15/Ti
Fuel assembly: wrapper	T91

2.3. Materials selection for ALLEGRO

ALLEGRO is thought of as an experimental reactor with a power of around 80 MW(th), with the objectives of demonstrating the viability and qualifying specific GFR technologies such as the fuel, fuel elements and specific safety systems. Moreover, in the area of materials and technologies, it is important to take advantage of the synergies with VHTR development. Indeed, the helium coolant and concepts for energy conversion using a gas turbine are common to both systems so materials and component research carried out for VHTR is largely relevant to GFR [18]. The main elements and operational conditions of the ALLEGRO system are shown in Fig. 4.

Recently, the consortium developing ALLEGRO has foreseen a re-design of the secondary circuit to take account of the findings calculated for a complete station black-out. These modifications envisage, among others, the addition of a He turbine and a He compressor; the air cooler becomes a He to air heat exchanger and the MHX becomes a He–He heat exchanger. As shown in Fig. 4, the operational core inlet and outlet temperatures of ALLEGRO are relatively low and therefore conventional steels are considered for this system. However, for an industrial prototype the temperatures would be above 350°C and other materials, e.g. 9Cr ferritic/martensitic (F/M) steel for the vessel, Ni-alloys for the heat exchanger and SiC/SiC for the core components are envisaged.

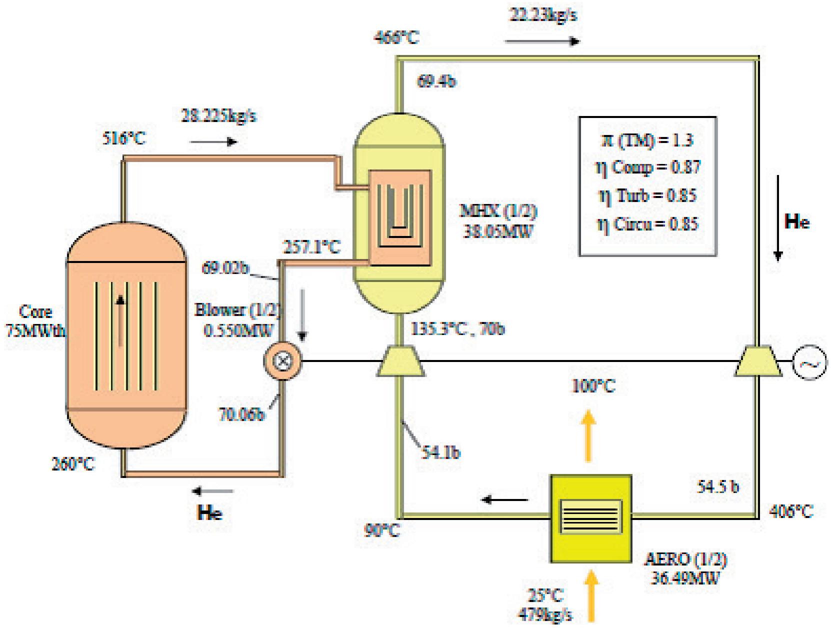


FIG. 4. Layout and operational parameters of ALLEGRO [19].

3. EUROPEAN EXPERIMENTAL PROGRAMMES FOR MATERIALS DEVELOPMENT AND CHARACTERIZATION

As described in the previous section, the ESNII facilities are systems that are developed in the medium term in order to demonstrate innovative technological solutions and to perform irradiation experiments on innovative fuel and materials. For these reasons, the reference structural and core materials selected for these systems are conventional steels. However, for further development of the fast neutron systems, to reach the declared targets of safety combined with sustainability and economic competitiveness, innovative materials are needed. In particular, innovation is required for the core components such as the fuel cladding. On the basis of these considerations, the EERA JPNM has been structured in the following four subprogrammes (SP) [6]:

— SP1: Materials R&D to support ESNII.
— SP2: Innovative steels development, e.g. ODS.

— SP3: High temperature ceramic and refractory materials development, e.g. SiC_f/SiC.
— SP4: Multiscale modelling and validation experiments

3.1. Materials R&D to support ESNII

In this subprogramme, the objective is to perform pre-normative research activities mainly on AISI 316LN and 9Cr F/M steels. Most of the activities foreseen are included in the EURATOM supported project MATTER [5]. Although a wide database is available on these two classes of steels, there are some items that need further investigation.

As far as the AISI 316LN steel is concerned, this steel has good creep properties and good corrosion resistance in conventional environments (e.g. water/steam and liquid Na) and at moderate temperatures. Therefore, it is considered as reference material for, for instance, the vessel of LFR and SFR concepts. Experiences with AISI 316LN steel in the FR environment are available. However, it is relevant to future systems for the performance of this steel to be assessed for an in-service time of at least 60 years. Innovation over the past 60 years regarding this steel, should be taken into account with respect to nuclear applications. For this purpose it is of relevance to deepen the fundamental understanding of the degradation mechanism that can occur on bare and welded AISI 316LN when it is in service. In this context, long term creep, creep-fatigue and long term ageing tests in the FR environment (for the ESNII facilities, mainly liquid Na and liquid Pb/LBE) are relevant for the long operational time assessment.

The 9Cr F/M steel is under consideration for different applications, e.g. heat exchanger, steam generator and wrapper tube in the case of SFR and LFR, cladding in the case of LFR and ADS, and vessel in the case of GFR. A reason for such a wide variety of applications might be due to the fact that F/M steels, already used in heat exchangers of ‘conventional’ power plants, have a higher thermal conductivity and a lower thermal expansion with respect to austenitic steels. This can be an advantage, for instance for the heat exchanger design, by reducing needed materials and therefore costs. However, there are several mechanical properties of 9Cr F/M steels that need to be carefully analysed, e.g. the softening during cyclic in-service operation conditions (see Fig. 5 from Ref. [20]), the resistance to corrosion and to environmental assisted mechanical properties degradation in steam and in liquid metals, the creep-to-rupture strength for cladding applications (see Fig. 6 from Ref. [21]) and the welding of thick components as vessels.

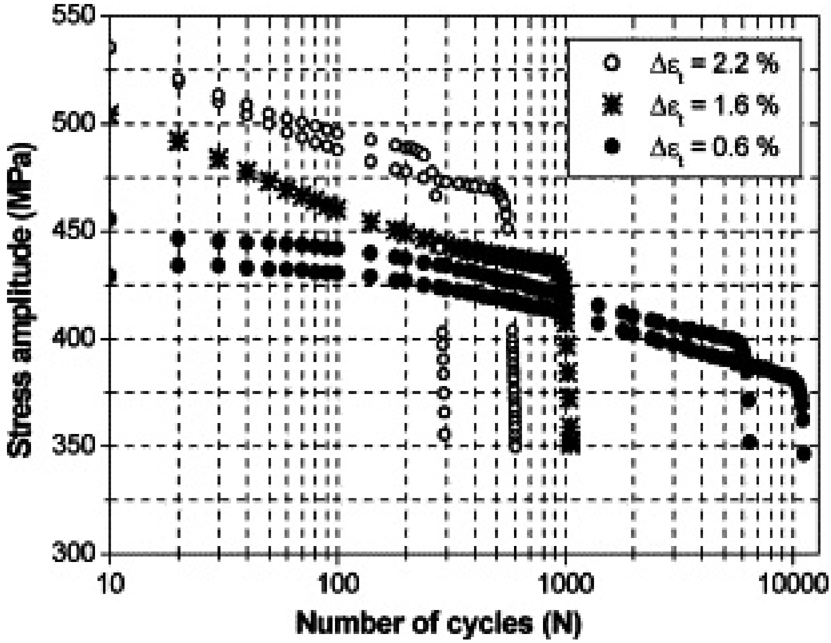


FIG. 5. Evolution of the stress amplitude versus the number of cycles in air at 300°C at different strain amplitudes of the T91 steel tempered at 750°C. Martensitic steels such as T91 exhibit a cyclic softening very early in the fatigue life. Then, at the end, a marked decrease of the stress amplitude occurs and is related to the propagation of a macroscopic crack into the bulk before final failure [20].

3.2. Innovative steels development

A key objective of fast neutron systems is to increase the fuel burnup to ~20at.% so as to improve the preservation of resources and the high burnup might also be of relevance for the transmutation of minor actinides (and in some context also Pu) for waste management purposes. In this context, the development of an appropriate cladding material is of paramount importance. As discussed in Ref. [22], the envisaged high fuel burnup would translate into an irradiation dose at the cladding higher than 150 dpa. For such irradiation doses the 15Cr-15Ni Ti stabilized austenitic steel, which is the reference clad material developed in the past European FR programmes, shows important swelling and concurrent hardening/embrittlement [23]. Therefore, the options to overcome this inconvenience are either to continue to improve the austenitic steel [24] or to change to steel matrix less prone to swelling [22].

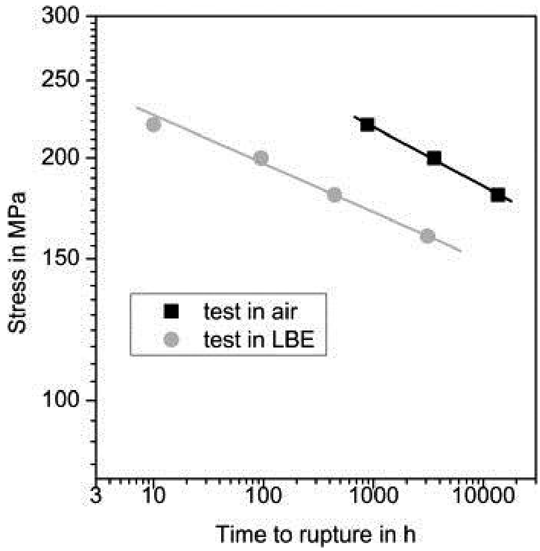


FIG. 6. Stress versus rupture time of T91 steel in air and LBE. The tests have been performed at 550°C. The creep-rupture time of bare T91 in LBE is considerably reduced when compared to air [21].

One option that has been envisaged in Europe is to develop and improve ODS (oxide dispersed strengthened) F/M alloys [3, 6]. As discussed in Ref. [22], F/M steels show very low dimensional changes when irradiated up to ~150 dpa. However, their high temperature strength limits their application to temperatures lower than 550°C (wrapper in SFR and LFR). By strengthening the F/M steel matrix through dispersion of nanoscale oxide particles, the high temperature strength can be improved. The development line for ODS steels is on the one hand the martensitic ODS alloys and on the other ferritic ODS steels. For both lines advantages and disadvantages can be envisaged [22]. The main challenges for these materials are: the development of a reproducible and cost efficient production route, the resolution of anisotropic effects (see Fig. 7 from Ref. [22]) and the high temperature stability. Moreover, a full qualification programme is needed before the use of ODS as fuel cladding. This programme should include screening tests with different combinations of temperature, neutron fluence and fuel; further tests are: experiments with artificially defective pins for safety purposes; assessment of bundle behaviour in real dimensions; tests of one or more fuel assemblies with the same specification to confirm statistics of experimental results; tests under operational and mild overpower transients, such as at reactor start-up or at load-follow operation and off-normal operational conditions and tests simulating accident conditions, e.g. in the case of core disruption. Finally,

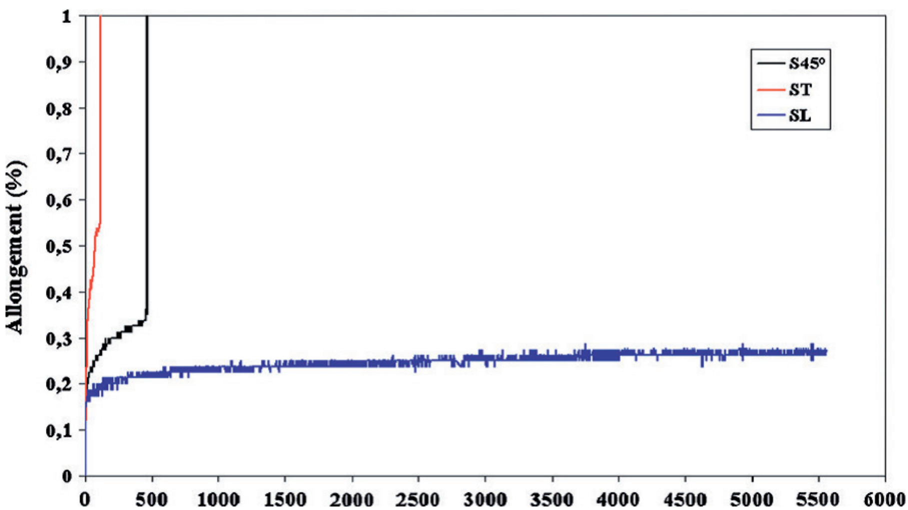


FIG. 7. Thermal creep curves of 18 Cr ODS alloy elaborated by hot extrusion as sheet. Tests have been performed at 650°C and 300 MPa. ODS ferritic steels exhibit a strong anisotropy in creep behaviour which results in rupture times shorter in the transverse direction than in the longitudinal direction and creep rates lower in the longitudinal direction over other directions. The longitudinal properties are very favourable compared with the transverse direction [22].

parts of a core should be loaded with a selected subassembly to show validity of the concept before a complete loading of a core to confirm validity of the concept. Moreover, compatibility with fuel, coolant and with re-processing approaches (e.g. Ref. [25]) should also be confirmed.

3.3. High temperature ceramic and refractory materials development (e.g. SiC_f/SiC)

In the case of very high temperature applications, as foreseen in the GFR core where operational temperatures up to ~1200°C are envisaged, steels and even ODS steels would not be applicable as clad materials. Moreover, owing to their neutron properties (capture), refractory materials, such as W, Mo, etc., cannot be considered. Therefore, ceramic composites such as the SiC_f/SiC (SiC fibres embedded in a SiC matrix) are under discussion. Promising SiC_f/SiC claddings are under development at the CEA [26]. As for all cladding materials, also in the case of SiC_f/SiC, the gas tightness, the high mechanical strength and the high thermal conductivity are key features that need to be satisfied. Moreover, this material is also considered as the SFR's wrapper, whose properties at high temperature could confer advantages on core safety.

As reported in Ref. [27], SiC_f/SiC has several advantageous features, e.g. good thermochemical stability as well as neutron transparency; SiC_f/SiC shows stable mechanical properties up to 1000°C and good stability under irradiation if appropriate fibres are selected. However, SiC_f/SiC is a material which can undergo damages in the elastic range, changing properties, e.g. leaktightness and thermal conductivity, when subjected to mechanical stresses and to irradiation.

As far as SiC_f/SiC development is concerned, it is of high relevance that appropriate fabrication procedures are selected for the desired geometry and properties [26]. Moreover, gastightness has been addressed through the design of a 'sandwich' cladding, which has an intermediate metallic liner [27] (see Fig. 8 from Ref. [27]). The R&D programme should include the assessment and possibly the optimization of fabrication routes to reach the objective of demonstrating cladding ease of fabrication. In particular, parameters such as thermal conductivity, mechanical properties, joining process and tightness route are considered for the optimization of the fabrication process. Finally, the optimal fabrication route will be selected by taking into account such parameters as cost, ease of fabrication in the final form with respect to design tolerance, joining, coating capability, etc.

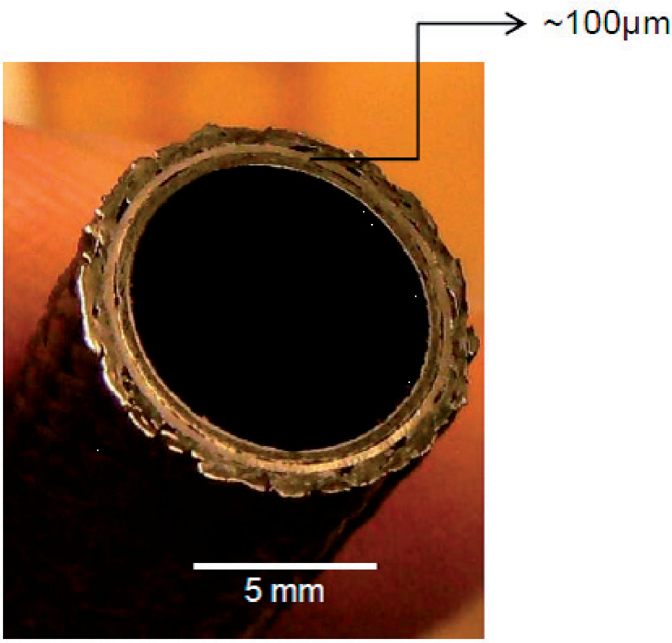


FIG. 8. Examples of sandwich cladding prototype SiC/SiC + metallic liner + SiC/SiC [27].

The screening tests of selected composites should occur in the GFR representative environment and should include high temperature mechanical resistance and corrosion/erosion resistance in He containing oxidizing impurities. The high temperature mechanical tests should be performed at very high temperatures with appropriate equipment. Further investigations concern the fuel/liner compatibility and the mechanical behaviour under irradiation. In particular, ion irradiation tests can be used to screen the composites in terms of radiation resistance and to perform first basic studies.

3.4. Multiscale modelling

As discussed in the previous sections, materials for innovative nuclear systems will have to withstand the combination of high energy neutron irradiation up to very high doses, high operating temperatures, pressures and thermal gradients, and aggressive coolants. Under these conditions, a number of phenomena that severely affect the integrity of materials are known to appear. For instance, for temperatures below ~400°C, it is known that irradiation hardening and embrittlement is a problem for conventional steels. The combination of higher irradiation temperatures and doses can cause materials to change their volume and shape due to swelling and irradiation creep, as observed on austenitic steels. Moreover, the contact with aggressive coolants (liquid metals such as Pb and LBE, gases, and also water, or supercritical water) can cause potentially not only corrosion and erosion but also degradation of the mechanical performance of the material.

All of these phenomena are almost never the consequence of only one factor and will most certainly influence each other. Experiments under conditions partially simulating those expected in future reactors are in certain cases possible, but they are long and expensive. Moreover, facilities in which neutron irradiation experiments can be carried out are very limited in number not only in Europe but worldwide. In this situation, extrapolations guided by understanding of the physical mechanisms that drive the response of materials under the conditions to which they will be subjected can be of very valuable support to the material's development.

As described in Ref. [28], the European programme on modelling is composed of two parts, i.e. short-to-medium term activities, with the objective of supporting the ESNII facilities development. In this framework, a blend of both semi-empirical and physics based models, together with suitable modelling oriented experiments involving in-depth materials characterization is used. The experimental activities should provide indications on the variables influencing a specific problem (e.g. embrittlement, creep, swelling, for a given category of material) and on the fundamental mechanisms that determine it. Moreover, finite element approaches or trend curves that provide approximate quantification for specific problems (as above) and tools based on more fundamental information (e.g. Monte Carlo type) are also developed. The long term activities should provide physics based modelling. These are meant to progress steadily along the path leading to the development of sophisticated, fully physics based models, to be eventually integrated and chained in multiscale common platforms.

4. SUMMARY AND OUTLOOK

In the European framework, an initiative has been started with the objectives of creating and developing synergies between the different national programmes and their facilities, and over the long term to bundle them in a unique European Joint Program on Nuclear Materials (JPNM) that addresses materials development, characterization and validation for innovative fast neutron reactors. The added value of the JPNM can be measured through the synergetic effects and the identification of common objectives. The JPNM has at present four subprogrammes with the focus of supporting the ESNII facilities and developing innovative materials that will be essential for the operation of industrial fast neutron facilities. In particular, the R&D activities include the pre-normative tests on conventional reference steels such as the AISI 316L and the 9Cr F/M steels. Moreover, ODS and SiC_f/SiC are also at the core of the JPNM. Finally, the multiscale modelling is considered a key to supporting the development of materials and their qualification. The JPNM is evolving and there is a new approach to include in the experimental and theoretical approaches to both fuel–clad interaction items, as well as theoretical and experimental activities related to the development of innovative fuels.

ACKNOWLEDGEMENTS

The authors wish to thank all colleagues working in the framework of the JPNM and the two EU projects GETMAT and MATTER.

REFERENCES

[1] SUSTAINABLE NUCLEAR ENERGY TECHNOLOGY PLATFORM, Research and Innovation Agenda (SRIA) 2013
www.snetp.eu

[2] SUSTAINABLE NUCLEAR ENERGY TECHNOLOGY PLATFORM, The European Sustainable Nuclear Industrial Initiative Concept Paper
http://www.snetp.eu/esnii/

[3] FAZIO, C., et al., European cross-cutting research on structural materials for Generation IV and transmutation systems, J. Nucl. Mater **392** (2009) 316.

[4] FAZIO, C., et al., Innovative materials for Gen IV systems and transmutation facilities: The cross-cutting research project GETMAT, J. Nucl. Eng. Des. **241** (2011) 3514.

[5] LEBARBÉ, T., et al., “Presentation of FP7 MATTER project: General overview”, Proc. American Society of Mechanical Engineers ASME Pressure Vessels and Piping Conference, ASME, Baltimore, MD, 2011 (2011).

[6] FAZIO, C., et al. “European energy Research Alliance Joint Program Nuclear Materials”, Proc. European Energy Conference 2012, Maastricht, The Netherlands (2012).

[7] EUROPEAN ENERGY RESEARCH ALLIANCE, Nuclear Materials,
http://www.eera-jpnm.eu/

[8] GAUCHE, F. and ROUAULT, J., French SFR R&D Program and Design Activities for SFR Prototype ASTRID, Energy Procedia **7** (2011) 314.

[9] ALEMBERTI, A., “The ALFRED project on Lead-cooled Fast Reactor”, Proc. ESNII Conference Advanced fission research in Horizon 2020, European Economic Social Committee, Brussels, Belgium (2012).

[10] STAINSBY, R., et al., Gas cooled fast reactor research in Europe, J. Nucl. Eng. Des. **241** (2011) 348.

[11] ZAETTA, A., “The French prototype of 4th generation reactor ASTRID”, Proc. WE Heraeus Seminar, Bad Honnef, Germany, 2011 (2011).

[12] GELINEAU, O., “Key properties requirements for ASTRID materials” Deliverable D1.1, Materials Testing and Rules (MATTER) Project, June 20, 2011.

[13] VASILE, A., Astrid Advanced Sodium Technological Reactor for Industrial Demonstration, IAEA Technical Working Group on Fast Reactors (TWG-FR), Activity Report for the Period May 2011–May 2012, Argonne Natl Lab., IL (2012).

[14] ABDERRAHIM, H.A., et al., MYRRHA — A multi-purpose fast spectrum research reactor, J. Energy Convers. Manag. **63** (2012) 4.

[15] STUDIENCENTRUM VOOR KERNENERGIE (SCK. CEN), MYRRHA project,
http://myrrha.sckcen.be/en/RD/Material_qualification

[16] SCHUURMANS, P., “MYRRHA system configuration”, Proc. Int. Workshop on Innovative Nuclear Reactors Cooled by Heavy Liquid Metals: Status and Perspectives, Pisa, Italy, 2012 (2012).

[17] ALEMBERTI, A., “ELFR-DEMO: ALFRED System Configuration”, Proc. Int. Workshop on Innovative Nuclear Reactors Cooled by Heavy Liquid Metals: Status and Perspectives, Pisa, Italy, 2012 (2012).

[18] STAINSBY, R., et al., Gas cooled fast reactor research in Europe, J. Nucl. Eng. Des. **241** (2011) 3481.

[19] HORVÁTH, A., "ALLEGRO — a gas-cooled Fast Reactor demonstrator status of the project", Proc. ESNII Conference Advanced fission research in Horizon 2020, European Economic and Social Committee, Brussels, Belgium, 2012 (2012).

[20] VOGT, J. B., et al., Mechanical behaviour of the T91 martensitic steel under monotonic and cyclic loadings in liquid metals, J. Nucl. Mater. **335** (2004) 222.

[21] JIANU, A., et al., Creep-to-rupture tests of T91 steel in flowing Pb–Bi eutectic melt at 550 °C, J. Nucl. Mater. **394** (2009) 102.

[22] DUBUISSON, Ph., et al., ODS Ferritic/Martensitic alloys for Sodium Fast Reactor fuel pin cladding, J. Nucl. Mater. **428** (2012) 6.

[23] DUBUISSON, Ph., "GEN IV Sodium Fast Reactors - 2. Materials challenges supporting new Sodium Fast Reactor designs", 1st International School on Materials for Nuclear Reactors, MATRE-1 Rochehaut, Belgium (2009).

[24] SCHÄFER, L., An austenitic steel remaining ductile after neutron irradiation, J. Nucl. Mater **231** (1996) 63.

[25] GWINNER, B., et al., Impact of the use of the ferritic/martensitic ODS steels cladding on the fuel reprocessing PUREX process, J. Nucl. Mater. **428** (2012) 110.

[26] DAVID, P., et al., "SiCSiC composities for GFR fuel cladding" CLEFS CEA No. 59, Summer 2010.

[27] LE FLEM, M., "Advanced Materials for Fuel Cladding in Sodium Fast Reactors (and an insight into Lead Fast Reactors)", Int. Workshop on Innovative Nuclear Reactors Cooled by Heavy Liquid Metals: Status and Perspectives, Pisa, Italy, 2012.

[28] MATERBA, L., "SUB-PROGRAMME 4: Modelling: correlations, simulation and experimental validation", EERA Joint Program Nuclear Materials, 2012.

MATERIALS OPTIONS FOR THE STEAM GENERATOR OF SODIUM COOLED FAST REACTORS

X.G. FU, B. LONG, L.Q. HAN, B. QIN, J.Q. ZHANG, S.X. WANG
China Institute of Atomic Energy,
Beijing, China

Abstract

Ferritic steels are the most preferred materials for steam generator (SG) applications in sodium cooled fast reactors (SFRs) due to their high resistance to aqueous stress corrosion cracking and enough high strength properties. Among the ferritic steels, 2.25Cr-1Mo (T22), 2.25Cr-1.6W (T23) and Modified 9Cr-1Mo (T91) were considered as the structural materials for the SG. A series of tests (including corrosion tests in sodium, high temperature high pressure water and high temperature water vapour) on these materials was performed by the China Institute of Atomic Energy. The final option for structural material for SG must consider service life in the SFR. A method to evaluate the performance of the SG was developed by calculating the lifetime arrangement of heat transfer tubes in the superheater, while the other issues such as sodium corrosion, decarburization, steam corrosion and creep damage at service temperature had to be taken into account. The analysis result shows that T22 steel is suitable as an SG structural material when the design lifetime of the reactor is 20–30 years, while T23 steel is suitable for 40–50 years and T91 has most advantages when the lifetime is extended to 50–60 years.

1. INTRODUCTION

Alloys 2.25Cr-1Mo, 2.25Cr-1.6W and Modified 9Cr-1Mo were considered as the preferred structural materials to use for the steam generator (SG) in sodium cooled reactors (SFRs). 2.25Cr-1Mo steel has successfully performed when used as evaporheater or superheater for a number of years. Modified 9Cr-1Mo steel is a promising candidate material for the once-through type of SG. Having a strength approaching and even exceeding some high chromium steels, 2.25Cr-1.6W steel is also considered for the SG. The three types of steel have been used extensively in fossil fuel power plants. Some fossil fuel boilers have similar service conditions as the SFR, such as steam oxidation and degradation of mechanical properties [1–3].

Service life in the SFR plays an important role in the choice of SG structural material. In this paper, the lifetime of heat transfer tubes in the superheater is calculated to evaluate the performance of these steels. Under operation conditions, the heat transfer tubes contact with sodium and water/steam at elevated temperature. The degradation of the heat transfer tube was defined by three issues as follows: (i) creep damage at high temperature, (ii) wall thickness thinning due to sodium corrosion and steam oxidation, (iii) mechanical properties degradation affected by thermal ageing and decarburization in sodium [4–6]. This paper presents one simple method for estimating the service life of the heat transfer tube. Some calculated data in this paper come from a series of experiments carried by the China Institute of Atomic Energy (CIAE) with Baosteel Co. Ltd. The other is based on the long term service results from fossil power plants considering similar service conditions of boilers and SGs in SFRs. The purpose of this paper is to give a service life assessment of heat transfer tubes made of 2.25Cr-1Mo, 2.25Cr-1.6W and modified 9Cr-1Mo steel for use in the SG.

2. METHOD

The performance of the SG was evaluated using a mean diameter formula and life fraction model, as expressed as Eqs 1 and 2 [7, 8]. The heat transfer tube of Φ 16 × 2.5 mm was evaluated for an SG operating at nominal steam conditions of 773 K and 15 MPa in an SFR. Based on the period of chemical cleaning in the SG, the service life was divided into many cycles of 10 000 h to evaluate the working capacity of the heat transfer tube in terms of the absence of fracture on the cross-section. If the calculation result does not meet the requirement of Eq. 1 or 2, fracture on the cross-section occurs and the performance of materials approaches the end of service life.

$$\sigma = P\left(\frac{D_o}{2S} - 0.5\right) \leq [\sigma] \tag{1}$$

where σ is the stress (MPa), P is the pressure force (MPa), D_o is the outside diameter (mm) and S is the wall thickness (mm), [σ] is the allowable stress (MPa). The stress on the cross-section is calculated considering wall thickness thinning due to sodium corrosion and steam oxidation at each cycle. The allowable stress is the minimum value of yield strength adjusted by a factor of 1.5 and ultimate tensile strength by a factor of 2.6. A reduction factor due to decarburization is given for 2.25Cr-1Mo and 2.25Cr-1.6W steels.

$$D_{LF} = \sum t_i / t_{ri} \leq 1 \tag{2}$$

where D_{LF} is the damage factor, t_i is the time at any temperature any stress and is the time to rupture at that temperature and stress. The life fraction model which is often identified as Robinson's rule was selected to evaluate the creep damage of the heat transfer tube. The t_{ri} at each cycle was determined by a curve of stress against creep rupture time based on a Z-parameter method proposed by Zhao et al. The Z-parameter method considers the scatter of creep rupture data and proposes a reliable curve for creep rupture prediction based on a curve of stress against the Larson-Miller Parameter of the steel.The detail of the Z-parameter method is described in Refs [9–12].

3. THINNING OF WALL THICKNESS

3.1. Sodium corrosion

The corrosion tests were conducted in a static sodium furnace with a thermal convection loop from 683 K to 873 K for a duration varying from 3000 h to 5000 h. Test materials included 2.25Cr-1Mo, modified 9Cr-1Mo, austenitic steel and weld samples. The oxygen level in the sodium was lowered to 25 ppm at the beginning of each test and the corrosion rate was evaluated by the dissolution of alloy elements into sodium. Test results indicated that no local corrosion was observed and the corrosion effect of sodium led to an inconsiderable thinning of wall thickness. As a result, the corrosion rate of the three steels in sodium is determined as 1 μm/year between 723 K to 798 K, as illustrated in Fig. 1.

3.2. Steam oxidation

Steam oxidation tests were carried out in a horizontal furnace equipped with a quartz tube. The material samples were exposed to pure steam at 723 K and 773 K for 250 h and periodically removed for examination. The test results in Fig. 2(a) show that 2.25Cr-1Mo and modified 9Cr-1Mo steel have the similar growth rate and structure of oxide scales. Steam oxidation tests for 2.25Cr-1.6W were not carried out, but the steel should have the same behaviour with regard to steam oxidation. It was surprising to find that the corrosion rate in steam is

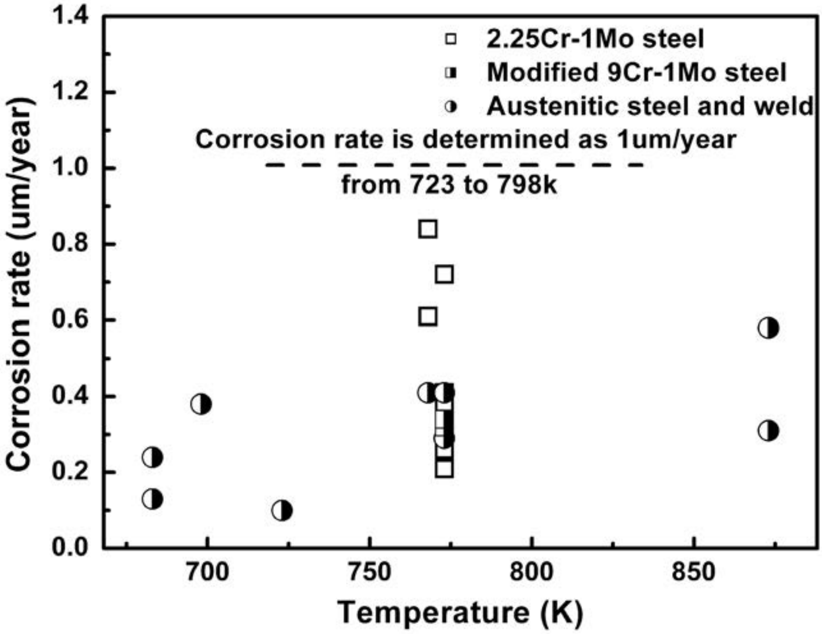


FIG. 1. Corrosion rate of some steels in sodium.

up to 330 μm/year based on these results. Figure 2(b) shows some long service results of steam oxidation in fossil power plants for these steels [13–16]. Comparing these results, it was found that scale thickness versus time for 2.25Cr-1Mo and modified 9Cr-1Mo steels obey the parabolic law and the slope of short term results is not suitable to allow extrapolation of an annual corrosion rate from the initial parabolic curves. Considering the whole oxide scale removed by cycle chemical cleaning, the data close to 10 000 h from 755 K to 844 K were used to determine the corrosion rate in steam at 773 K, as in Eq. 3. As a result, the corrosion rate of the three steels in steam was determined as 38 μm/year at 773K.

$$v = 33.475 + 2.16E - 14\exp(T/23.4544) \quad (3)$$

where ν is corrosion rate of the three steels (μm/year) and T is the temperature (K).

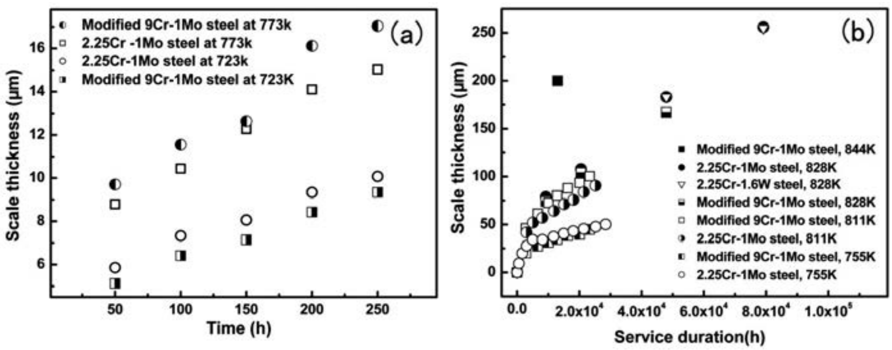


FIG. 2. Corrosion rate of three steels in steam.

4. DEGRADATION OF MECHANICAL PROPERTIES

4.1. Effect of thermal ageing

Thermal ageing was expected to reduce the tensile and creep rupture properties. Some data mentioned in the literature for these steels with long term service in fossil power plants were compiled to determine the mechanical properties with experimental results of original materials [15–29]. Figure 3(a) and (b) show the tensile properties of 2.25Cr-1Mo steel fell obviously in the early stage and began to reduce slightly after 50 000 h. The steel samples were removed for examination from main steam pipes operating at nominal steam conditions of 813 K and 10 MPa. The maximum reduction of strength for 2.25Cr-1Mo steel was about 35%. The reduction of tensile properties for 2.25Cr-1.6W and modified 9Cr-1Mo steels was about 15%. The steel samples were removed for investigation from the superheater or reheater operating at 825–866 K. The upper limit of allowable stress was determined as a uniform value for each cycle's calculation based on the reduction of yield strength (YS) and ultimate tensile strength, as illustrated in Fig. 3. For 2.25Cr-1Mo steel, the yield strength at 773 K was 174 MPa and the ultimate tensile strength at 773 K was 260 MPa, based on minimum value of service results. For 2.25Cr-1.6W and modified 9Cr-1Mo steels, reductions of 20% and 30% were applied to the original yield strength and ultimate tensile strength at 773 K, respectively. The original yield strength at 773 K was 414 MPa for 2.25Cr-1.6W and 422 MPa for modified 9Cr-1Mo steels. The original ultimate tensile strength at 773 K was 473 MPa for 2.25Cr-1.6W and 550 MPa for modified 9Cr-1Mo [30]. In addition, the lower limit of allowable stress for evaluation was determined as 64 MPa for 2.25Cr-1Mo, 111 MPa for 2.25Cr-1.6W and 131 MPa for modified 9Cr-1Mo steel, referring to ASME 2010 Section III, Division 1-NH and the T23/24 book edited by V&M [30–32].

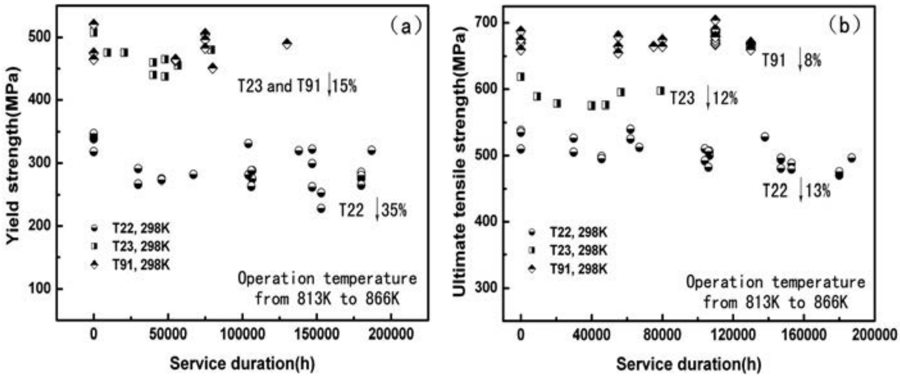


FIG. 3. The reduction of tensile properties at 298 K for three steels after exposure in fossil power plants. (a) Yield strength against service duration, (b) ultimate tensile strength against service duration.

Creep rupture properties were also affected by thermal ageing. Figure 4(a) illustrates the Larson-Miller parameter curves for 2.25Cr-1Mo steel at different service durations and the data were selected from creep rupture tests of these steels, which were conducted at the same temperature [17–24]. It was apparent that the time to rupture of 2.25Cr-1Mo steel reduced with increasing service time. In order to investigate the effect of long term thermal ageing, the Larson-Miller parameters for exposed 2.25Cr-1Mo steel compared with a master curve determined by the Z-parameter method based on the creep rupture data from NIMS Creep Sheet Date No. 3B [33] are illustrated in Fig. 4(b). It shows that the reduction of P values at high stresses is larger than that at low stresses and it also indicates the master cuvre will move to the left with increasing service time. According to Fig. 4(b), if the master curve is moved to the left by a certain P value based on service time, the reduction of P value at low stresses is larger than the actual P value. It means that the time to rupture at low stresses calculated by the Z-parameter method based on P values on the moving master are lower than the actual value and it represents a conservative method for estimating the degradation of creep rupture properties affected by long term thermal ageing. The t_{ri} value to evaluate the creep damage conditons at each cycle in Eq. 2 was determined by a moving master curve with a large reduction of P value. If the damage factor D_{LF} calculated in an entire service life is lower than 1, indicating that the heat transfer tubes have enough capacity to resist creep damage. Comparing the P value of 147 000 h exposed at 813 K and 140 MPa to the master curve, the result of time to rupture for 2.25Cr-1Mo steel is shown in Table 1 calculated by the Z-parameter method from the moving curve with a P value 1.14 reduction as illustrated in Fig. 4(b).

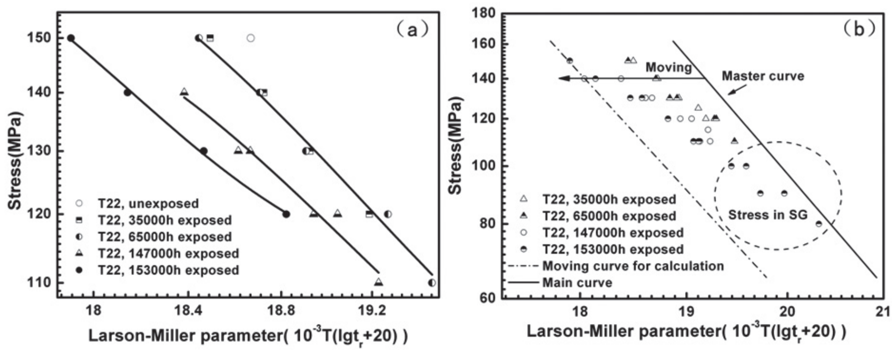


FIG. 4. The degradation of creep rupture properties for 2.25Cr-1Mo steel. (a) Larson-Miller parameter at different service duration, (b) Larson-Miller parameters compared with a master curve.

TABLE 1. RELIABILITY 99% OF Z-PARAMETER METHOD PREDICTION BASED ON THE MOVING CURVE OF 2.25CR-1Mo

Temperature (K)	Stress (MPa)	Serivce condition (h)	Time to rupture (h)	Prediction result (h)
813	140	147 000 exposed	153	74.5
813	107.87	153 000 exposed	1418.7	395.5
873	78.45	153 000 exposed	1879.0	72.6

As the exposed temperature is higher than 773 K and the pressure stress of the SG in the SFR is below 90 MPa, the moving curve determined above is conservative enough to evaluate the creep damage conditions for 2.25Cr-1Mo steel. In another example, the reduction of the P values for modified 9Cr-1Mo and 2.25Cr-1.6W steels were determined using a similar method (Fig. 5). The master curve for modified 9Cr-1Mo steel was determined by the Z-parameter method based on the creep rupture data from Haney et al. [34] and Baosteel. The master curve for 2.25Cr-1.6W steel was determined from the creep rupture data from Baosteel [35].

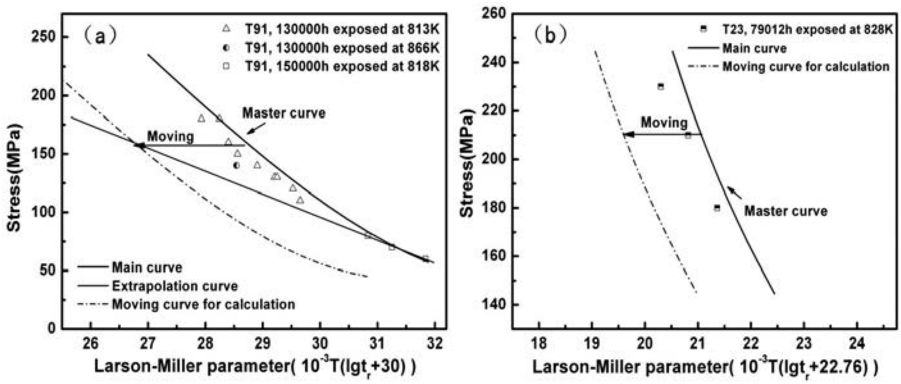


FIG. 5. Larson-Miller parameters comparing with a master curve. (a) Modified 9Cr-1Mo steel, (b) 2.25Cr-1.6W steel.

An extrapolation curve was illustrated in Fig. 5(a) based on the P values of 150 000 h exposed at 818 K of modified 9Cr-1Mo steels, which were the failure samples removed from a steam pipe in a fossil power plant [36]. This indicated the maximum reduction of P values for modified 9Cr-1Mo steel up to failure as the slope of the curve based on some actual P values decreasing with reduction of stress. As a result, a reduced P value of 1.96 for modified 9Cr-1Mo steel was determined by the difference between the master curve and the extrapolation curve at 160 MPa. As the data are limited, the reduced P value of 1.46 for 2.25Cr-1.6W steel was determined as six times the difference between the master curve and the P value at 210 MPa. Referring to service conditions of exposed T23 described in Fig. 5(b) and the reduced P value of modified 9Cr-1Mo steel, it was conservative to evaluate the creep damage conditions for 2.25Cr-1.6W steel.

4.2. Effect of decarburization in sodium

Figure 6 shows two typical profiles of carbon concentration against distance for 2.25Cr-1Mo steel in a sodium environment [37, 38]. For curve 1, the decarburization profile can be calculated from Fick's Law and shown in Eq. 4.

$$C(X) = C_0 + (C_s - C_0)\ \mathrm{erfc}\ (X/\sqrt{4Dt}) \tag{4}$$

where *C(X)* is the concentration at a distance *X* from the surface at the time *t*, C_0 is the initial concentration in the solid, C_s is the surface concentration and *D* is the diffusivity coefficient of carbon in metal.

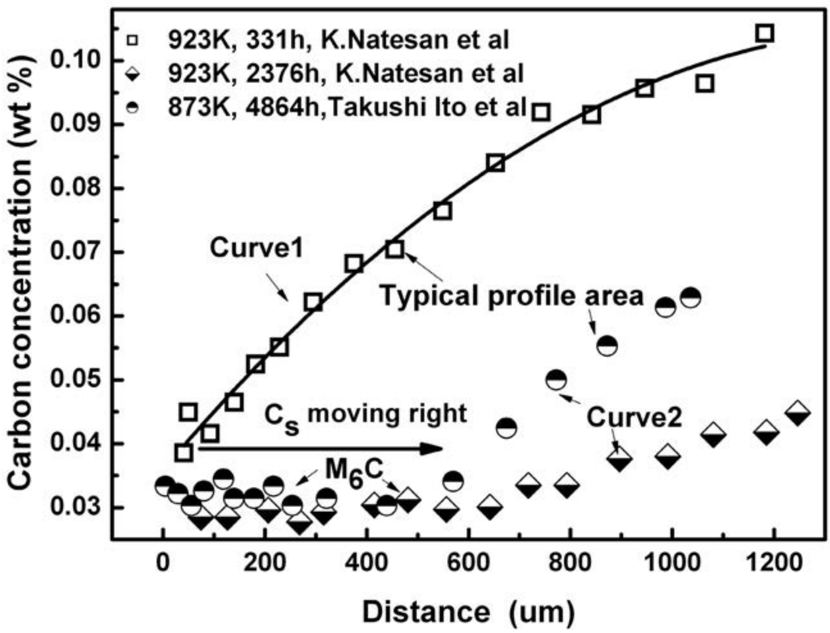


FIG. 6. Carbon concentration versus distance profiles in 2.25Cr-1Mo steel.

However, curve 2 exhibits a significant deviation, which is reported by Natesan et al. and Takushi et al., which could not be calculated by Fick's Law. The difference of two typical profiles may be attributed to the transformation of carbide. It seems that most of the carbon diffusing to the surface is produced by transformation of carbide in the typical profile area. The diffusion of carbon in this area also obeys Fick's Law. However, in the flat profile area the M_6C is stable and no further transformation of carbide occurs. Consequently, the length of decarburization could be predicted by determining the migration rate of C_s, with the assumption that C_s has a uniform motion to the right. In addition, the migration rate of C_s at various temperatures is approximately proportional to the diffusion coefficient of carbon in metal at corresponding temperatures, as illustrated in Table 2 [37–39].

TABLE 2. THE RELATIONSHIP OF C_S MIGRATION RATE AND CARBON DIFFUSION COEFFICIENT

Temperature (K)	Flat profile Length (μm)	Exposed time (h)	Cs migration rate (μm/h)	Carbon diffusion coefficient (cm^2/s)
923	750	2376	0.316	1.5E-9
873	500	4864	0.103	5.63E-10
811	127	8000	0.016	5.69E-11

Based on the diffusion coefficient of carbon of 1.87E-11 cm^2/s for 2.25Cr-1Mo steel, the migration rate of C_s at 773 K was determined as 30 μm/year. For 2.25Cr-1W steel, the migration rate of C_s is assumed as 15 μm/year with consideration of carbon stabilized by addition elements. As a result, the length of decarburization was defined as the length of the flat profile and it could be determined by the migration rate of C_s and the time for each cycle's calculation. Finally, a reduction factor for 2.25Cr-1Mo and 2.25Cr-1W steels was estimated from Eq. 5.

$$R = 1 - \frac{0.3h}{S} \tag{5}$$

where h is the length of decarburization and S is the wall thickness of tube at each cycle. The maximum reduction of 30% is for complete carbon loss through wall thickness. The modified 9Cr-1Mo steel is expected to minimize the extent of decarburization with no mechanical strength reduction [38].

5. CALCULATION RESULT AND CONCLUSION

Based on the results discussed in the previous section, the data used for service life calculation at each cycle are summarized in Table 3.

TABLE 3. THE DATA USED FOR SERVICE LIFE CALCULATION AT EACH CYCLE

		T22	T23	T91
Sodium corrosion rate (μm/year)		1	1	1
Steam oxidation rate (μm/year)		38	38	38
The lower limit of allowable stress (MPa) (ASME Code and T23/T24 book)		64	111	131
The upper limit of allowable stress	Yield strength (MPa) (reduction of 30%)	174[a]	290	296
	Yield strength (MPa) (reduction of 20%)		331	337
	Ultimate tensile strength (MPa) (reduction of 30%)	260[a]	331	385
	Ultimate tensile strength (MPa) (reduction of 20%)		378	440
Rate of decarburization length change (μm/year)		30	15	0

[a] The upper limit of allowable stress for 2.25Cr-1Mo just used 174 MPa yield strength and 260 MPa ultimate tensile strength determined in previous section for calculation.

The evaluation of three steels was calculated from Eqs 1 and 2 at each cycle up to the absence of fracture on the cross-section. Table 4 shows the calculation result for a lifetime for a heat transfer tube made of 2.25Cr-1Mo, 2.25Cr-1.6W and modified 9Cr-1Mo steel in the SG. The result indicates that the 2.25Cr-1Mo steel is suitable to be the SG structural material when the design lifetime of reactor is 20–30 years, while 2.25Cr-1.6W steel is for 40–50 years and modified 9Cr-1Mo steel has most advantages when the lifetime is extended to 50–60 years.

TABLE 4. THE CALCULATION RESULT OF LIFETIME FOR THREE TYPES OF STEEL IN SG

	Parameter	The lower limit of allowable stress	The upper limit of allowable stress		Lifetime (capacity utilization factor = 0.8)
			Reduction of 30%	Reduction of 20%	
T22	Service life D_{LF}	150 000 h 0.228	230 000 h[a] 0.846		21–32 years
T23	Service life D_{LF}	300 000 h 0.231	320 000 h 0.295	340 000 h 0.432	42–48 years
T91	Service life D_{L_F}	360 000 h 0.146	380 000 h 0.399	390 000 h 0.707	51–55 years

[a] The upper limit of allowable stress for 2.25Cr-1Mo was 174 MPa yield strength and 260 MPa ultimate tensile strength as determined in the previous section.

ACKNOWLEDGEMENTS

The authors wish to thank Prof. Zhao Jie, Prof. Wang Qijiang and Dr. Tsoshio OGATA for supplying data and for the invaluable discussion.

REFERENCES

[1] CHETAL, S.C. and VAIDYANATHAN, G., “Evolution of Design of Steam Generator for Sodium Cooled Reactors”, Proc. Of 3rd Itl. Conf. on Heat Exchangers, Boilers and Pressure Vessels (HEB-97), Alexandria, Egypt, 1997, Atomic Energy Establishment, Cairo, Egypt (1997) Vol. 1.

[2] Management group research and development. R&D report 1992-In support of the European Fast Reactor EFR.

[3] KLUEH, R.L., Elevated-temperature ferritic and martensitic steels and their application to future nuclear reactors, Oak Ridge National Laboratory, Rep. ORNL/TM, Oak Ridge, TN (2004)176.

[4] MANNAN, S.L, CHETAL, S.C, BALDEV, R. and BHOJE, S.B., Selection of materials for prototype fast breeder reactor, Indira Gandhi Centre for Atomic Research (ICGAR), Kalpakkam, India (2004).
[5] GORYNIN, I. V., et al., Design and choice of materials for fast reactors with sodium heat-transfer agent, J. Metal Sci. Heat Treat. International **41** (1999) 389–395.
[6] NATESAN, K., MEIMEI, L., CHOPRA, O.K. and MAJUMDAR, S., Sodium effects on mechanical performance and consideration in high temperature structural design for advanced reactors. J. Nucl. Mater. **392** (2009) 243–248.
[7] KANEMARU, O., et al., Remaining life prediction by the iso-stress method of boiler tubes after prolonged service, J. Mater. Sci. Res. International **3** (1997) 31–36.
[8] SWINDEMAN, R.W., et al., "Understanding damage mechanisms in ferritic/martensitic steels", Proc. Of 17th Annual Conf. on Fossil Energy Materials, Baltimore, MA, 2003, Oak Ridge Natl Lab., Oak Ridge, TN (2003).
[9] ZHAO, J., HAN, S.Q., et al., Remaining life assessment of a CrMoV steel usingthe Z-parameter method, International J. Press. Vessels Pip. **81** (2004) 757–760.
[10] ZHAO, J., et al., Statistical analysis and reliability prediction of creep rupture property for T91/P91 steel, Acta Metallurgica Sinica. **45** (2009) 835–839 (in Chinese).
[11] FENG, W., ZHAO, J., et al., Reliability analysis of rupture strength of a steel based on Z parameter, J. Metal Heat Treat. **32** (2007) 469–472 (in Chinese).
[12] ZING, L., Z-Parameter Method for Life Prediction and its Application of Heat-resistant Steels. MS thesis (in Chinese).
[13] GRIESS, J.C., MAXWELL, W.A., The Long-Term Oxidation of Selected Alloys in Superheated Steam at 482 and 538°C, Oak Ridge National Laboratory, Rep. ORNL-5771, Oak Ridge, TN (1981).
[14] ZHONG, X., et al., The characteristic of oxide scales on T91 tube after long-term service in an ultra-supercritical coal power plant, J. Supercrit. Fluids **72** (2012) 68–77.
[15] KOMAI, N., et al., 10-Year experience with T23(2.25Cr-1.6W) and T122(12Cr-0.4Mo-2W) in a power boiler, American Society of Mechanical Engineers, Transactions of the ASME, **127** (2005) 190–196.
[16] MASUYAMA, F., et al., 3-Year experience with 2.25Cr-1.6W (HCM2S) and 12Cr-0.4Mo-2W (HCM12A) steel tubes in a power boiler, JSME International J. **41** (1998) 1098–1104.
[17] CHEN, H.B., Experimental study on 10CrMo910 main steam pipe-line after 60,000h service, J. East China Electric Power **03** (1987) 44–46.
[18] BING, L., Variation of Mechanical Property and Microstructure for 10CrMo910 Main steam pipe steel during long-time service and the reliability of its operation, J. North China Electric Power, **03** (1991) 11-16.
[19] OU, Y.J., LI, J.F., et al., Life time assessment of 10CrMo910 main steam piping after 150,000h operation, J. Hebei Electric Power, China, **17** (1998) 40–46 (in Chinese).
[20] WANG, Z.W., CHEN, L.G., et al., Estimate of structure states and capabilities for 10CrMo910 main steam pipes after 147 thousand hour running, Engineering journal of Wuhan University, (2005) (in Chinese).

[21] CHEN, L.G., WANG, Z.W., et al., Evaluation of microstructure property for 10CrMo910 main steam piping No.1 of unit No.4 in Niangziguan Power plant, Thermal Power Generation, China, **04** (2004) 60–62 (in Chinese).

[22] SI, B.X., LIN, Z.Y., et al., Discussion on service life of 10CrMo910 Thin-walled main steam line, j. Zhe Jiang Electric Power **2** (2007) 55-58.

[23] QIAN, X.M., JIA, J.M., et al., Residual service assessment of 10CrMo910 main steam pipe-line in thermal power plant under ANQING petrochemical company, J. Thermal Power Generation, China **07** (2005) 40–56.

[24] SHEN, H.S., LIN, Z.Y., et al. Study on Varying Metallic Properties of Straight and Elbow Pipes of Main Steam Piping after Long-Time Service, J. Electricity **2** 4 (2005) 34–37.

[25] JIN, K., NIE, M., et al., Performance evaluation of T23 steel tube aging, j. Guang Dong Electric Power **20** 2 (2007) 44–47 (in Chinese).

[26] LIU, S.T., SHI, Z.G., et al., Leakage analysis of T23 steel high temperature reheater tube, PTCA Part: A Phys. Test. **45** 10 (2009) 638–644 (in Chinese).

[27] SWINDEMAN, R.W., SIKKA, V. K., et al., EVALUATION OF T91 AFTER 130,000 HOURS IN SERVICE ORNL, in Proc. American Society for Mechanical Engineers (ASME) 1998 Pressure Vessels and Piping Conference, San Diego, CA, (1998).

[28] SHI, Z.G., HOU, A.Z., et al., Changes in Mechanical Property and Microstructure of T91 steel During Service, J. Power Engineering China **05** 5 (2005) 742–746 (in Chinese).

[29] KE, H., ZHANG, X.Y., et al., Performance evaluation on T91 steel of high temperature reheater after long-time service, j. Heat Treatment of Metals **36** (2011) 127–131 (in Chinese).

[30] WANG Q., ZOU, F., ZHANG, K., R&D of T23 Hi-pressure Boiler Tube, J. Steel Pipe **34** 3 (2005) 16–19 (in Chinese).

[31] AMERICAN SOCIETY OF MECHANICAL ENGINEERS, "Class 1 Components in Elevated Temperature Service", ASME Boiler and Pressure Vessel Code, 2012 Edition, Section III, Division 1-Subsection NH, American Society of Mechanical Engineers, New York, NY (2010).

[32] ARNDT, J., HAARMANN, K., KOTTMAN, G., VAILLANT, J.C., BENDICK, W., KUBLA, G., et al. The T23/T24 book. Edition V&M, 2000.

[33] NATIONAL INSTITUTE FOR MATERIALS SCIENCE (NIMS), Structural Materials Data Sheet Online, Creep Sheet No.3B, 1986.

[34] HANEY, E.M., et al., Macroscopic results of long-term creep on a modified 9Cr-1Mo steel (T91), J. Mater. Sci. Eng. A **510-511** (2009) 99-103.

[35] WANG, Q., ZOU, F. et al., Effect of T23 steel's microstructural evolution on its properties, J. Baosteel Technology **3** (2006) 18-22 (in Chinese).

[36] CHEN, Y., The research on microstructure and failure mechanism of T/P91 heat-resistant alloy steel after long-term service at elevated temperature. Master thesis (in Chinese).

[37] NATESAN, K. et al., Compatibility of Fe-21/4wt% Cr-1wt% Mo steel in a sodium environment, J. Nucl. Techn. **28** (1976) 441-451.

[38] ITO, T. et al., Evaluation of carburization and decarburization behavior of Fe-9Cr-Mo ferritic steels in sodium environment, J. Nucl. Sci. Techn. **29** 4 (1992) 367-377.

[39] D.W. KOCH, P.J. KOVACH, Materials Examination of a Model Sodium Heated Steam Generator, AEC Contract at(II-I)-1280, BAW-1280-37, Research report No.5131, (1965).

IAEA
International Atomic Energy Agency

No. 23

ORDERING LOCALLY

In the following countries, IAEA priced publications may be purchased from the sources listed below or from major local booksellers.

Orders for unpriced publications should be made directly to the IAEA. The contact details are given at the end of this list.

AUSTRALIA
DA Information Services
648 Whitehorse Road, Mitcham, VIC 3132, AUSTRALIA
Telephone: +61 3 9210 7777 • Fax: +61 3 9210 7788
Email: books@dadirect.com.au • Web site: http://www.dadirect.com.au

BELGIUM
Jean de Lannoy
Avenue du Roi 202, 1190 Brussels, BELGIUM
Telephone: +32 2 5384 308 • Fax: +32 2 5380 841
Email: jean.de.lannoy@euronet.be • Web site: http://www.jean-de-lannoy.be

CANADA
Renouf Publishing Co. Ltd.
5369 Canotek Road, Ottawa, ON K1J 9J3, CANADA
Telephone: +1 613 745 2665 • Fax: +1 643 745 7660
Email: order@renoufbooks.com • Web site: http://www.renoufbooks.com

Bernan Associates
4501 Forbes Blvd., Suite 200, Lanham, MD 20706-4391, USA
Telephone: +1 800 865 3457 • Fax: +1 800 865 3450
Email: orders@bernan.com • Web site: http://www.bernan.com

CZECH REPUBLIC
Suweco CZ, spol. S.r.o.
Klecakova 347, 180 21 Prague 9, CZECH REPUBLIC
Telephone: +420 242 459 202 • Fax: +420 242 459 203
Email: nakup@suweco.cz • Web site: http://www.suweco.cz

FINLAND
Akateeminen Kirjakauppa
PO Box 128 (Keskuskatu 1), 00101 Helsinki, FINLAND
Telephone: +358 9 121 41 • Fax: +358 9 121 4450
Email: akatilaus@akateeminen.com • Web site: http://www.akateeminen.com

FRANCE
Form-Edit
5 rue Janssen, PO Box 25, 75921 Paris CEDEX, FRANCE
Telephone: +33 1 42 01 49 49 • Fax: +33 1 42 01 90 90
Email: fabien.boucard@formedit.fr • Web site: http://www.formedit.fr

Lavoisier SAS
14 rue de Provigny, 94236 Cachan CEDEX, FRANCE
Telephone: +33 1 47 40 67 00 • Fax: +33 1 47 40 67 02
Email: livres@lavoisier.fr • Web site: http://www.lavoisier.fr

L'Appel du livre
99 rue de Charonne, 75011 Paris, FRANCE
Telephone: +33 1 43 07 50 80 • Fax: +33 1 43 07 50 80
Email: livres@appeldulivre.fr • Web site: http://www.appeldulivre.fr

GERMANY
Goethe Buchhandlung Teubig GmbH
Schweitzer Fachinformationen
Willstätterstrasse 15, 40549 Düsseldorf, GERMANY
Telephone: +49 (0) 211 49 8740 • Fax: +49 (0) 211 49 87428
Email: s.dehaan@schweitzer-online.de • Web site: http://www.goethebuch.de

HUNGARY
Librotrade Ltd., Book Import
PF 126, 1656 Budapest, HUNGARY
Telephone: +36 1 257 7777 • Fax: +36 1 257 7472
Email: books@librotrade.hu • Web site: http://www.librotrade.hu

INDIA

Allied Publishers
1st Floor, Dubash House, 15, J.N. Heredi Marg, Ballard Estate, Mumbai 400001, INDIA
Telephone: +91 22 2261 7926/27 • Fax: +91 22 2261 7928
Email: alliedpl@vsnl.com • Web site: http://www.alliedpublishers.com

Bookwell
3/79 Nirankari, Delhi 110009, INDIA
Telephone: +91 11 2760 1283/4536
Email: bkwell@nde.vsnl.net.in • Web site: http://www.bookwellindia.com

ITALY

Libreria Scientifica "AEIOU"
Via Vincenzo Maria Coronelli 6, 20146 Milan, ITALY
Telephone: +39 02 48 95 45 52 • Fax: +39 02 48 95 45 48
Email: info@libreriaaeiou.eu • Web site: http://www.libreriaaeiou.eu

JAPAN

Maruzen Co., Ltd.
1-9-18 Kaigan, Minato-ku, Tokyo 105-0022, JAPAN
Telephone: +81 3 6367 6047 • Fax: +81 3 6367 6160
Email: journal@maruzen.co.jp • Web site: http://maruzen.co.jp

NETHERLANDS

Martinus Nijhoff International
Koraalrood 50, Postbus 1853, 2700 CZ Zoetermeer, NETHERLANDS
Telephone: +31 793 684 400 • Fax: +31 793 615 698
Email: info@nijhoff.nl • Web site: http://www.nijhoff.nl

SLOVENIA

Cankarjeva Zalozba dd
Kopitarjeva 2, 1515 Ljubljana, SLOVENIA
Telephone: +386 1 432 31 44 • Fax: +386 1 230 14 35
Email: import.books@cankarjeva-z.si • Web site: http://www.mladinska.com/cankarjeva_zalozba

SPAIN

Diaz de Santos, S.A.
Librerias Bookshop • Departamento de pedidos
Calle Albasanz 2, esquina Hermanos Garcia Noblejas 21, 28037 Madrid, SPAIN
Telephone: +34 917 43 48 90 • Fax: +34 917 43 4023
Email: compras@diazdesantos.es • Web site: http://www.diazdesantos.es

UNITED KINGDOM

The Stationery Office Ltd. (TSO)
PO Box 29, Norwich, Norfolk, NR3 1PD, UNITED KINGDOM
Telephone: +44 870 600 5552
Email (orders): books.orders@tso.co.uk • (enquiries): book.enquiries@tso.co.uk • Web site: http://www.tso.co.uk

UNITED STATES OF AMERICA

Bernan Associates
4501 Forbes Blvd., Suite 200, Lanham, MD 20706-4391, USA
Telephone: +1 800 865 3457 • Fax: +1 800 865 3450
Email: orders@bernan.com • Web site: http://www.bernan.com

Renouf Publishing Co. Ltd.
812 Proctor Avenue, Ogdensburg, NY 13669, USA
Telephone: +1 888 551 7470 • Fax: +1 888 551 7471
Email: orders@renoufbooks.com • Web site: http://www.renoufbooks.com

United Nations
300 East 42nd Street, IN-919J, New York, NY 1001, USA
Telephone: +1 212 963 8302 • Fax: 1 212 963 3489
Email: publications@un.org • Web site: http://www.unp.un.org

Orders for both priced and unpriced publications may be addressed directly to:

IAEA Publishing Section, Marketing and Sales Unit, International Atomic Energy Agency
Vienna International Centre, PO Box 100, 1400 Vienna, Austria
Telephone: +43 1 2600 22529 or 22488 • Fax: +43 1 2600 29302
Email: sales.publications@iaea.org • Web site: http://www.iaea.org/books

15-01091

Following on from the success of FR09, which was held in Kyoto in 2009, this conference — FR13 — was attended by some 700 experts from 27 countries and 4 international organizations representing different fields of fast reactor and related fuel cycle technologies. The programme comprised 41 technical sessions organized into 10 topical tracks.

The conference provided a forum to exchange information and experience on national and international programmes and new developments in fast reactors and related fuel cycle technologies, with a focus on safe technologies and sustainable scenarios. The papers in these Proceedings have been peer reviewed by the members of the International Scientific Programme Committee and include invited papers and contributed papers (on CD-ROM).

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA
ISBN 978-92-0-104114-2
ISSN 0074-1884