Fast Reactors and Related Fuel Cycles: Safe Technologies and Sustainable Scenarios

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

VOLUME 2

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EDITED BY S. MONTI

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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.

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CONCEPT AND DEVELOPMENT STATUS OF FAST BREEDER REACTOR FUELS IN THE FACT PROJECT

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Abstract

A conceptual design study and related R&D on the JSFR (Japan sodium cooled fast reactor) with mixed oxide (MOX) fuels, advanced aqueous reprocessing and simplified pelletizing fuel fabrication as a promising concept have been implemented in the fast reactor cycle technology development (FaCT) project. The fuel concept is being established in the FaCT project to improve economic potential in the fuel cycle and to enhance safety characteristics. Ferritic core materials and large diameter fuel pins with annular pellets will be adopted in the high burnup fuel. An inner duct will be equipped in the fuel subassembly to mitigate the core disruptive event. To actualize the concept, key technologies to be developed are oxide dispersion strengthened (ODS) ferritic steel with high temperature mechanical strength for fuel pin claddings and the simplified pelletizing fuel fabrication system, including a microwave de-nitration of plutonium enrichment adjusted solution and die wall lubrication. In the present status of the project, many basic technical findings of ODS ferritic steel have been obtained in the field of powder metallurgy, mechanical properties and irradiation characteristics. The application potential of the simplified pelletizing method has been confirmed. Furthermore, the properties of MOX fuel bearing minor actinides (MAs), including melting point and thermal conductivity, have been systematically measured to develop the MA-bearing MOX fuel with the aim of reducing the amount and the toxicity of radioactive wastes. The design technology of the MA-bearing MOX fuel with annular pellets has been also studied.

1. INTRODUCTION

The fast breeder reactor (FBR) and its fuel cycle technology will provide promising solutions for sustainable energy resource and environmental issues. As a result of the feasibility study, the system which combines the sodium cooled FBR with mixed oxide (MOX) fuels, advanced aqueous reprocessing and simplified pelletizing fuel fabrication was selected as the most promising concept that could meet design requirements and technical viability. The fast reactor cycle technology development (FaCT) was commenced in cooperation with the Ministry of Education, Culture, Sports, Science and Technology, the Ministry of Economy, Trade and Industry, the Federation of Electric Power Companies of Japan, the Japan Electrical Manufacturer's Association and the Japan Atomic Energy Agency (JAEA) [1].

In the first phase of the FaCT project, focused efforts were devoted to developing the selected key technologies of a system named the Japanese sodium cooled fast reactor (JSFR). The conceptual design study of the JSFR was promoted as well. Prospects for these key technologies were identified and various technical outcomes were provided.

However, the accident of the Fukushima Dai-ichi nuclear power station caused by the Great East Japan Earthquake in March 2011 has made a significant impact on the nuclear policy of Japan. Consequently, the FaCT project has been suspended but significant investigations have been conducted to enhance safety towards the establishment of safety design criteria of the sodium cooled FBR. An innovative strategy for energy and the environment was decided in September 2012 by the energy and environment council of the Japanese Government. In this strategy, it is reported that research and development aimed at the reduction of the amount and toxic level of radioactive wastes should be promoted. This policy will be taken into account in the future fuel development plan for fast reactors.

In this paper, the concept of fuel design and the typical achievements of fuel development in the project are reported.

2. FUEL DESIGN CONCEPT IN THE FACT PROJECT

To achieve targets including 'safety and reliability', 'sustainability', 'economic competitiveness', and 'nuclear non-proliferation', the conceptual design of the JSFR has been studied. Table 1 summarizes the main design specifications of the JSFR and its fuel. Figure 1 illustrates the schematic view of the fuel subassembly for the JSFR. To improve economic competitiveness, a long operational period is aimed for. To reduce the cost of the fuel cycle system, average discharged burnup is expected to be approximately 150 GW·d/t. Large diameter fuel pins are selected to improve the internal conversion rate by the large volume fraction of the fuel and to control burnup reactivity loss over the long operational period. High density annular pellets exhibit excellent thermal performance in terms of avoiding fuel melting and pellet cladding mechanical interaction resistance to high burnup. The maximum fast neutron dose is to be approximately 250 dpa for the high burnup core design. A coolant outlet temperature is set at 550°C. The maximum cladding temperature is required up to 700°C to achieve the high coolant outlet temperature for increased power discrepancy in the high burnup core. The oxide dispersion strengthened (ODS) ferritic steel is to be adopted as the cladding material to meet these challenging

TABLE 1. MAIN DESIGN SPECIFICATIONS OF THE JSFR AND ITS FUEL

	Low breeding option	High breeding option
Output thermal power (MWt)	3530	←
Output electric power (MWe)	1500	←
Cycle length (months)	26	21
Refueling batch [Core/RB]	4 / 4	←
Coolant inlet temp. (degrees C)	395	←
Coolant outlet temp. (degrees C)	550	←
Core height (cm)	100	75
Axial blanket region (cm) [Upper/Lower]	20 / 20	40 / 50
Fuel pin diameter (mm)	10.4	9.3
Cladding material	ODS	ODS
Number of fuel pins per subassembly	255	315
Pellet type	Annular	←
Pu-enrichment (wt%) [Inner/Outer]	18 / 21	22 / 24
MA-content (wt%) [core averaged]	1.1	1.2
Breeding ratio	1.1	1.2
Discharge burnup (GWd/t) [core]	approx. 150	←
[core + blanket]	approx. 90	approx. 60
Maximum neutron dose (dpa)	approx. 250	←
Maximum linear heat rate (W/cm)	approx. 410	approx. 420

requirements, such as neutron dose resistance and mechanical strength at high temperature.

To reduce the amount and potential radiotoxicity of radioactive wastes, minor actinide (MA) elements recovered in the advanced aqueous reprocessing method are to be recycled with plutonium and uranium elements. In this context, the MA-bearing MOX fuel is to be adopted for the JSFR's fuel. MA fission during irradiation in a core and the amount of MAs can be controlled. The influence of MAs on fuel properties and irradiation performance is being investigated to ensure the fuel integrity in fuel design.

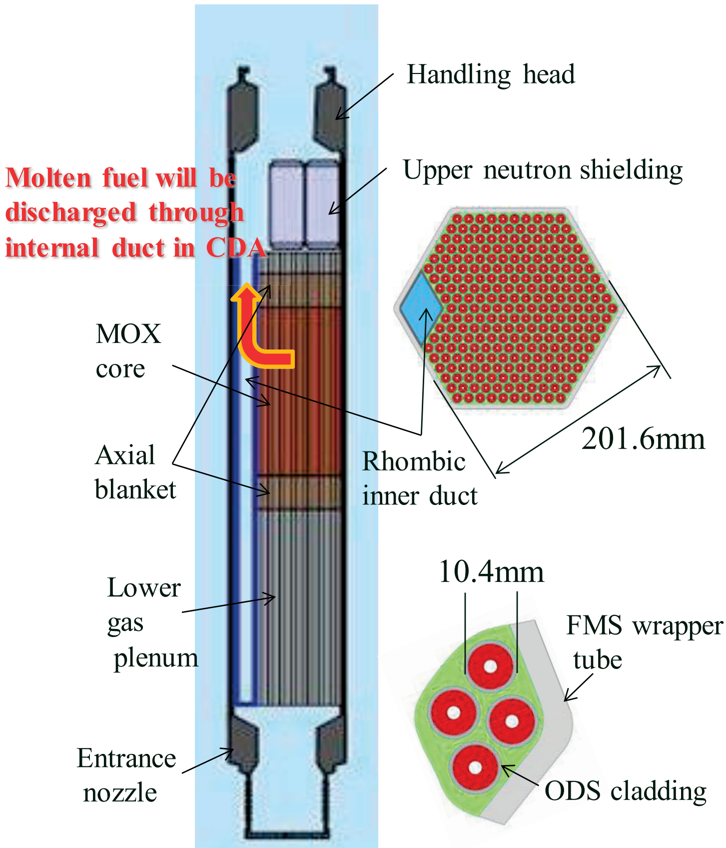


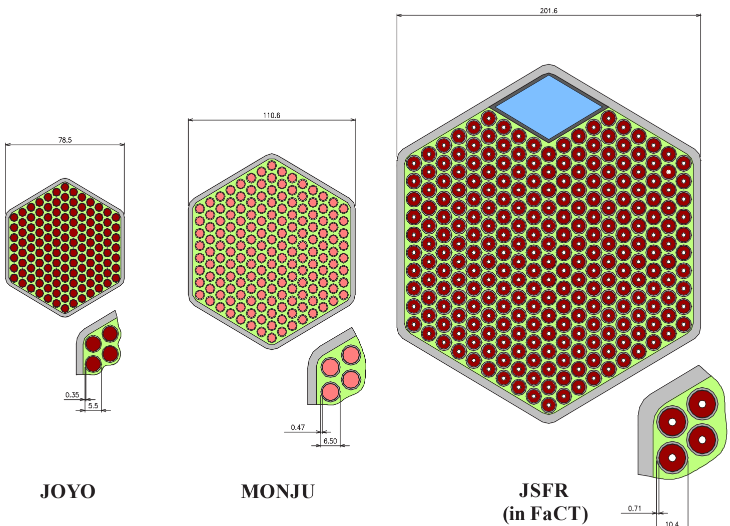
FIG. 1. Structure of fuel subassembly for JSFR.

As for a safety aspect, the FAIDUS (fuel assembly with inner duct structure) concept is adopted to avoid recriticality in the case of a core destructive accident, which is an accident category with extremely low probability as beyond the design basis events. In the early stage of a core destructive accident, molten fuel with pressurized fission gases will burst through an inner duct wall and will be discharged upward to prevent the formation of a large scale molten fuel pool within a core, which is one of factors leading to a severe power excursion. The inner duct is the device which has been newly introduced into the FBR fuel subassembly. In the beginning of its development, a joint method between the rhombic inner duct and a wrapper tube was studied through a small scale trial fabrication. The future irradiation experiment of a fuel subassembly with the inner duct is expected to confirm its in-pile integrity.

3. CURRENT STATUS OF FUEL DEVELOPMENT OF FACT

3.1. History of FBR fuel development at the JAEA

The JAEA developed MOX fuels for FBRs, step by step, towards the experimental reactor JOYO, the prototype reactor MONJU and JSFR, as shown in Fig. 2. The PNC316, which is cold worked austenitic steel modified minor constituent, is applied to the cladding and wrapper tube of the JOYO and MONJU fuel subassemblies. Various irradiation experiments have been conducted in JOYO to develop FBR fuels. These experiments have supplied considerable valuable data that have contributed to development of fuel design methods. Especially, fuel centerline temperature data obtained with the instrumented test assembly (INTA) equipped with thermocouples have enhanced the reliability of fuel thermal design methods. Some power-to-melt experiments were also conducted and they provided fuel performance data at a high linear heat rate. In a collaborative operational reliability testing programme between the US Department of Energy and PNC (the predecessor of the JAEA) [2], various



	JOYO	MONJU	JSFR (in FaCT)
Fuel Type	Solid/High density	Solid /Low density	Annular /High density
Fuel Density	94%T.D. (88%T.D.)*	85%T.D. (80%T.D.)*	(82%T.D.)*
Max. LHR	about 420 W/cm	about 360W/cm	about 430 W/cm
Burnup	90GWd/t (pin-average)	about 80GWd/t (discharged-average)	about 150GWd/t (discharged-average)
Cladding	PNC316	PNC316	ODS
Wrapper Tube	PNC316	PNC316	PNC-FMS (PH ferritic steel)

*Smear Density

FIG. 2. Overview of fuel subassemblies for JOYO, MONJU and JSFR.

irradiation experiments in the EBR-II were performed, not only in a steady state but also in off-normal conditions and these contributed greatly to the investigation into the irradiation behaviour of MOX fuels. On the basis of these experiences, technical development of the fuel for the JSFR was launched.

3.2. Development of core materials

The JAEA has developed ODS ferritic steel resistant to both a fast neutron dose of 250 dpa and a temperature of 700°C for fuel pin claddings and PNC-FMS (11Cr ferritic-martensitic steel) for wrapper tubes used at lower temperatures than claddings, as shown in Fig. 3 [3]. ODS ferritic steel, whose matrix is ferritic steel of good swell resistance, is improved in high temperature mechanical strength by addition of yttrium sesquioxide (Y_2O_3) dispersoids. ODS ferritic steel has been verified as possessing the necessary strength and providing enough elongation in tensile tests [4, 5].

Irradiation tests of 12 fuel pins with ODS ferritic steel claddings have been carried out at the Russian BOR-60 in order to obtain fundamental properties of

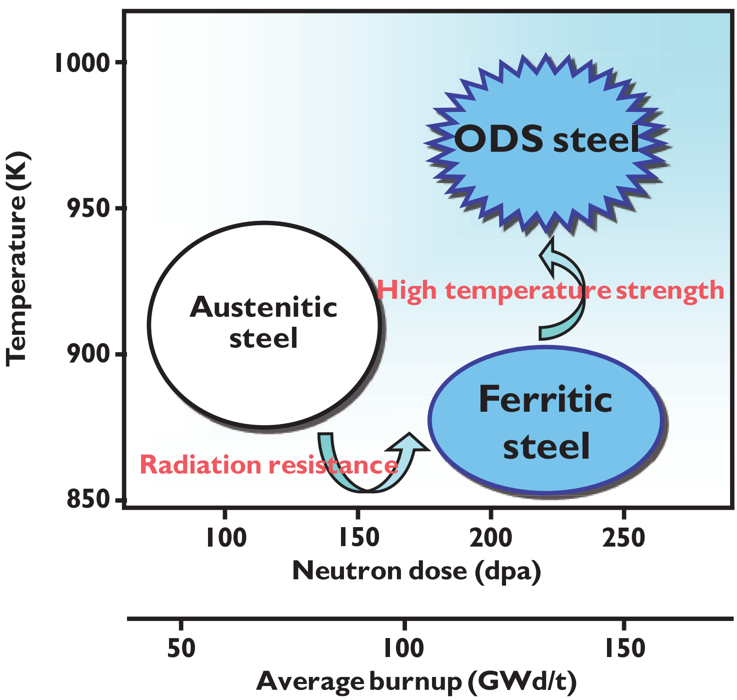


FIG. 3. Target of ODS ferritic steel for cladding tubes and ferritic steel for duct tubes.

the pins in collaboration with the Research Institute of Atomic Reactors, Russian Federation [6, 7]. The pins were irradiated up to the burnup of 112 GW·d/t and corresponding fast neutron dose of 51 dpa. Consequently, the cladding thickness reduction owing to inner surface corrosion was ~35 μm at the maximum and the in-pile integrity of a joint between an upper end plug and cladding by pressurized resistance welding was verified. However, one case of pin failure was observed, which might be ascribed to the heterogeneity of chemical compositions and the faulty defect of the cladding tube, and exposure to higher temperatures than the designed value. The development of cladding production by the pre-alloyed steel powder method, which enhances chemical homogeneity, is in progress.

3.3. Studies on fuel material properties

The JSFR is supposed to charge MOX fuel in its basic policy. Besides, in order to reduce the potential radiotoxicity of radioactive wastes, MAs, such as americium and neptunium, are added to MOX fuel. Therefore, the thermal and mechanical properties of the MA-bearing MOX, such as melting point, thermal conductivity and thermal expansion, are systematically studied at the JAEA [8].

The measurement of the melting point of MOX with the improved thermal arrest method which uses an inner rhenium container was carried out at the JAEA. The dependence of melting point on americium content has been obtained. The measurement reveals that the melting point of MA-bearing MOX changes continuously with americium content and its value is estimated with the ideal solution model calculation [9, 10]. The thermal conductivity of MA-bearing MOX is affected slightly with the americium content of 3wt%, except in the low temperature region [11], and is hardly affected with the neptunium content of 12wt% [12].

As basic data to study the material properties of MA-bearing fuel, the evaluation of (Pu, Am)O_2 properties, such as oxygen potential, diffusion coefficient of oxygen, phase diagram and thermal expansion, have progressed [13, 14].

3.4. Fuel design technology development

The annular fuel pellet is beneficial for fast reactors because of its availability for both high power and high burnup. The CEPTAR (computation code to evaluate fuel pin stability for annular fuel design) code is under development to evaluate the irradiation performance of a fuel pin with annular pellets [15]. In this code, the radial profile of fuel density is determined as a result of the migration of as-fabricated void distribution in a pellet, and subsequently, the central hole diameter is determined in accordance with the law of conservation of mass. Furthermore, central hole shrinkage is computed on the basis of the compress stress load around the central hole. Various irradiation experiments in JOYO, EBR-II and the prototype fast reactor (PFR) provided many fuel restructuring data, i.e. the diameters of central holes and columnar grain regions. The adequacy of the fuel restructuring model of the code is confirmed by the comparison between computed results and these observed data, including central hole shrinkage of annular pellets irradiated in the PFR up to 22at.%. The fuel temperature evaluation model is verified by fuel centerline temperatures obtained from INTA experiments in JOYO and power-to-melt data in JOYO, EBR-II and the material test reactor (MTR) at Harwell Laboratory [15].

It is required, as one of design criteria, that fuel melting should be prevented. The melting point of MOX is affected by the plutonium and americium contents. The migration model for plutonium and americium redistributions due to thermal diffusion and vapour phase transport via pores is introduced into the CEPTAR code. The accumulation of plutonium and americium around the central hole is observed by EPMA in the B11 short term irradiation test which was performed at JOYO in 2006 to investigate the thermal behaviour of MA-bearing MOX fuels at the beginning of irradiation [16]. In the B11 experiment, fuel pins with 2wt% americium- and 2wt% neptunium-bearing MOX pellets were irradiated at the maximum linear power of ~43 kW/m. It is confirmed that computation results by the CEPTAR code are in good agreement with the radial profiles of the plutonium and americium contents for (U, Np, Pu, Am)O_{2-x} fuels [17].

3.5. Development of fuel fabrication technology

An advanced pellet fabrication process is under development in the FaCT project to reduce the fabrication cost and to handle the decay heat and radioactivity of MOX fuel, especially that of MA-bearing MOX fuel. Figure 4 shows the current pellet fabrication method and the simplified pelletizing fabrication method for comparison. The simplified pelletizing fabrication method decreases significantly the number of processing steps. The following advantages are expected in this advanced method.

(i) The handling processes of raw powder, including powder mixing, are eliminated by the adjustment of Pu enrichment in a liquid state. The removal of these processes, in which the raw powder is scattered, can substantially reduce the worker's radiation exposure and radioactive wastes.

(ii) Following processes that are rationalized by minimization of organic additives in the granulation and pelletizing processes, the deterioration of additive by heat accumulation in the source powder, which may damage the quality of products, can be prevented.

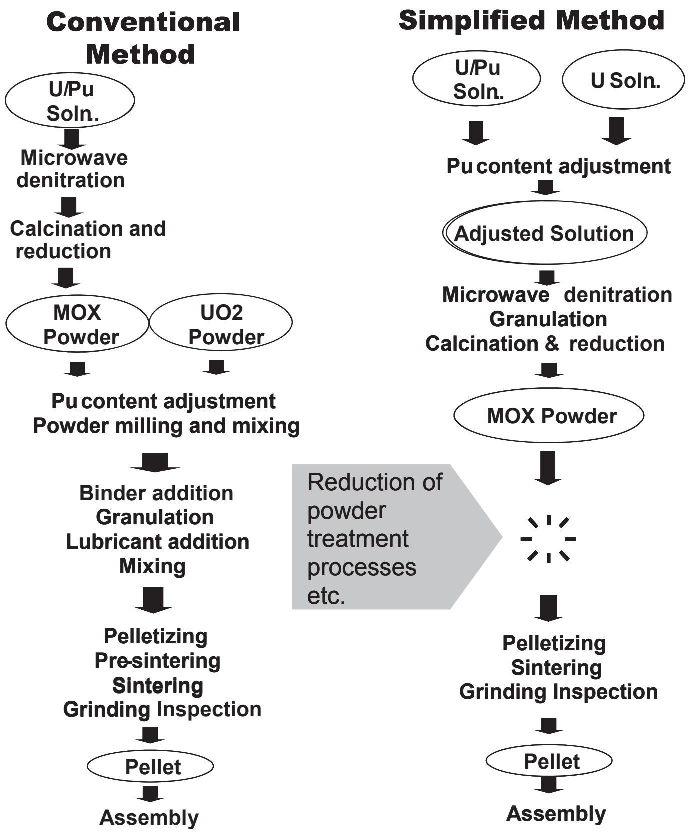


FIG. 4. The simplified pelletizing method comparing with conventional method.

Taking into account these advantages, the simplified pelletizing method can become an optimum one for remote handling of highly radioactive raw materials such as MA-bearing MOX fuel. Several innovative tasks are used to confirm the possibility of the proposed fuel production process, maintenance by remote operation and mass production ability. The main achievements of these tasks on the simplified pelletizing method are as follows [18, 19].

(1) Unified technology of conversion and granulation [20]

The unified process of de-nitration conversion of Pu/U mixed nitrate solution by microwave heating and granulation of MOX powder is under development to decrease powder scattering and to improve productivity. In the trial production, desirable MOX powder could be successfully provided with a specially designed agitation granulator in which an impeller could access the

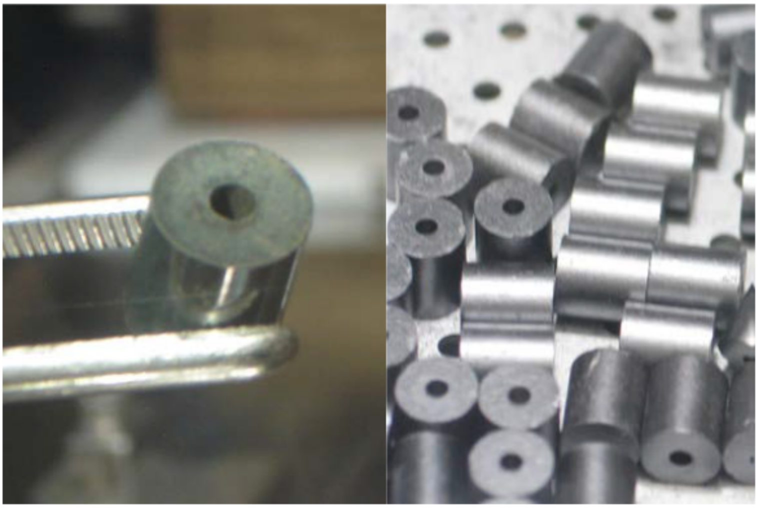
MOX bulk from above. The consecutive process of de-nitration, conversion and granulation in the same container was also demonstrated.

(2) Die wall lubrication pelletizing technology

The lubricant agent is sprayed on the die wall instead of being mixed into the source powder in the die wall lubrication pelletizing method. The way of controlling the amount of applied lubricant agent and the pelletizing characteristics of MOX powder have been investigated. The die wall lubrication pelletizing technology has been developed from a laboratory scale to a large scale. Annular pellets up to grade could be provided on a large scale MOX fabrication test, as shown in Fig. 5.

(3) Sintering and O/M ratio adjustment

To control the cladding internal corrosion, especially of the ODS cladding up to high burnup, the O/M ratio of MOX pellets is required to be below 1.97. A small scale furnace, in which the O/M ratio of MOX pellets could be adjusted, was designed and built in consideration of the results of fundamental experiments. It was confirmed by sintering experiments that the predetermined O/M ratio of MOX pellets with adequate density could be adjusted, and that it took additional



Green pellet Sintered pellet

FIG. 5. Pellets obtained from the large scale MOX test of die wall lubrication.

time to lower the O/M ratio to the targeted degree. Further investigations are expected to develop the O/M ratio adjustment technology.

4. CONCLUSION

The conceptual design study on the JSFR with MOX fuels has progressed in the FaCT project. Large diameter fuel pins with annular pellets and ODS ferritic steel claddings, which yield a high internal conversion rate, are to be adopted as the fuel concept suitable for high burnup to improve economic potential in the fuel cycle. The MA-bearing MOX fuel is applied to reduce the amount and the radiotoxicity of radioactive wastes. An inner duct, through which molten fuel is discharged to prevent recriticality in the case of a core destructive accident, is newly introduced and equipped within a fuel subassembly.

The fuel development in the FaCT project progressed on the basis of outcomes at JOYO and MONJU. In addition, the development of key technologies of the project, such as ODS ferritic steel cladding and the simplified pelletizing method, etc., advanced. Many findings of the ODS ferritic steel cladding in the technical fields of powder metallurgy, mechanical properties and irradiation characteristics have been obtained through trial production, out-of-pile tests and fuel pin irradiation experiments in BOR-60. The fundamental technologies for the unified process of both de-nitration conversion of Pu/U mixed nitrate solution by microwave heating and granulation of MOX powder, die wall lubrication pelletizing and O/M ratio adjustment were confirmed. Consequently, the groundwork for the simplified pelletizing method was established. Fuel properties, such as melting point, thermal conductivity, etc., of MA-bearing MOX were systematically investigated to develop the MA-bearing MOX fuels. In addition, the fuel design code is under development for the fuel pin with annular pellets made from MA-bearing MOX.

The accident at the Fukushima Dai-ichi nuclear power station, caused by the Great East Japan Earthquake in March 2011, has made a significant impact on the nuclear policy of Japan. Consequently, the FaCT project has been suspended. In fast reactor development in Japan, the subjects of radioactive waste and safety issues are still expected to be highlighted. Many of the technical outcomes achieved in the project will be of use for future development.

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DEVELOPMENT, FABRICATION AND CHARACTERIZATION OF FUELS FOR THE INDIAN FAST REACTOR PROGRAMME

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Abstract

The Indian Fast Reactor Programme started with the 40 MW(e) FBTR which was commissioned in October 1985 at Kalpakkam. The FBTR was fuelled with Mark I mixed carbide fuel of composition $(U_{0.3}Pu_{0.7})C$ for the initial core which was followed by Mark II fuel $(U_{0.45}Pu_{0.55})C$ for the extended core. The mixed carbide fuel has performed exceedingly well and its burnup has exceeded 1.65 GW·d/Te without any fuel failure. The fissile material (Pu) from spent fuel is being recovered gradually by reprocessing and is being recycled, thereby closing the carbide fuel cycle, which is an important milestone in the fast reactor programme. UO_2-45%PuO_2 fuel has also been loaded into the FBTR, making the core a hybrid with mixed carbide and mixed oxide fuel subassemblies. Construction of the 500 MW(e) Prototype Fast Breeder Reactor (PFBR) is at an advanced stage and manufacture of mixed oxide fuel pins for the PFBR is being done at Advanced Fuel Fabrication Facility at Bhabha Atomic Research Centre. R&D related to development of fast reactor fuels based on metallic fuels has been started. This consists of development of the technology for metallic fuel fabrication, building a database on the thermophysical properties of various fuel alloys, studies on thermodynamics and fuel–clad compatibility. Both mechanical and sodium bonded fuel designs are under consideration. A detailed study on thermophysical properties of binary U-15%Pu, which will be the fuel for mechanically bonded design, has been carried out. Work has been initiated on U based CERMET fuels for fast reactors, which has the potential for achieving a higher breeding ratio. U-15%UO_2 and U-30%UO_2 CERMET fuel pellets were prepared on an experimental scale by the powder metallurgy route which showed attractive features such as high thermal conductivity and good thermal shock resistance. In the future, work will be extended for development of U-PuO_2 CERMET fuel for fast reactors.

1. INTRODUCTION

From the very beginning of India's nuclear power programme, great emphasis has been laid on the efficient use of its nuclear resources. India has only modest resources of uranium but vast resources of thorium. Hence, the road map of the Indian nuclear power programme has to ultimately exploit the full potential of uranium and thorium resources. The use of uranium in the once through mode will enable use of only 1% of the energy potential of the available uranium resource. Fast reactors can use effectively the energy in uranium by converting the fertile isotope ^{238}U into the fissile isotope ^{239}Pu which results in an increase in the energy potential of natural U by a factor of about 60 [1]. Fast reactors can contribute to reducing the environmental burden of the spent fuel, which further enhances the long term sustainability of nuclear energy. Therefore, fast reactors are essential for India, not only for its contribution to nuclear power but for extension of its modest resources [2]. India started the fuel development programme for fast reactors in the early 1970s and now has a mature technology base for fabrication of $(U,Pu)O_2$ and (U,Pu)C fuels. The Fast Breeder Test Reactor (FBTR) with mixed carbide fuel has been in operation since 1985 and has provided a platform for development of fuel and other materials for commercial fast reactors. The Prototype Fast Breeder Reactor (PFBR), fuelled with mixed oxide fuel, is likely to go critical in 2013. Of late, a programme for development of high breeding metallic fuel and CERMET fuel has been launched.

2. DEVELOPMENT OF FUELS FOR FAST REACTORS

Fast reactor fuels should have the capability to operate at high heat ratings and achieve high burnup. The need for a compact core with high heat ratings implies that the fuel pins be of small diameter and separated from one another by narrow coolant channels. In short, the fuel for a fast reactor should meet the following requirements [1, 2]:

(a) High fissile atom density and as few moderating atoms as possible;
(b) Good thermal conductivity;
(c) Good compatibility with fuel cladding and reactor coolant;
(d) Low swelling from fission products.

The concentration of fissile material, namely, plutonium in a fast reactor fuel is much higher than that in a thermal reactor and the fuel fabrication technologies for fast reactors are accordingly more challenging. This paper deals with experience in fabrication, quality control and characterization of a wide variety of fuels such as oxide, carbide, metal and CERMET for the Indian fast reactor programme.

2.1. Early work on uranium–plutonium mixed oxide (MOX) fuel

Of all the forms of fast reactor fuel, uranium–plutonium mixed oxides (MOX) have the advantage of being the most stable ones in air, obviating the need for any special gas atmosphere in the process plants, and they are the most widely used fuels in fast reactors worldwide. The work on oxide fuel for fast reactors was initiated in India in the early 1970s. The initial composition for the FBTR was MOX fuel containing 30% PuO_2 with UO_2 having an 85% enrichment in U^{235}. The fabrication flowsheet for this fuel was developed in Bhabha Atomic Research Centre using PuO_2 and natural UO_2. However, owing to unavailability of enriched UO_2, this fuel composition was later abandoned. To overcome this shortcoming, a new fuel composition of 76% PuO_2 with natural UO_2 was taken up for development and evaluation. However, this fuel composition was also not pursued intensely because it was found to be incompatible with the liquid sodium and it had a considerably lower thermal conductivity. This set the stage in the 1980s for development of plutonium–uranium mixed carbide fuel for the FBTR.

2.2. Fabrication of mixed carbide fuel for the FBTR

The FBTR at Kalpakkam (40 MW(th)) at IGCAR has now become the test bed for development of fuel, blanket and structural materials for the Indian fast breeder reactor programme. The FBTR achieved criticality on 18 October 1985 with a unique Pu rich mixed carbide fuel. The initial criticality of the FBTR was achieved with a small core containing $(Pu_{0.7}U_{0.3})C$ fuel (MK-I fuel) and the reactor was operated for several campaigns at low power levels, mainly to gain operational experience and to carry out various reactor physics and safety experiments. The small core was then progressively expanded with $(Pu_{0.55}U_{0.45})C$ fuel (MK-II fuel) to increase the electrical power level of the reactor for fuel. As there was no information available for these fuel compositions in the open literature, development of fuel fabrication flow heets, fuel pellet specifications, characterization techniques, generation of thermophysical property data and out-of-pile fuel–clad–coolant compatibility tests had to be carried out indigenously.

Mixed carbide fuel is prepared by powder metallurgy processes comprising carbo-thermic reduction of oxide powder in vacuum. The process involves co-milling of UO_2 and PuO_2 powder with graphite, followed by compaction of the milled powder at low pressure. This is done to increase the contact of oxide with graphite, leaving enough porosity for expulsion of the gaseous reaction product, CO. These pre-compacts are heated in a vacuum furnace at around 1500°C. The mixed carbide clinkers so formed are crushed and milled to obtain sinterable grade mixed carbide powder. The mixed carbide powder is mixed with a suitable binder/lubricant and then pre-compacted, granulated, compacted to green pellets and sintered in an Ar–H_2 gas mixture. Pellets are then stacked and loaded in the fuel cladding tube and welded by TIG under a helium atmosphere to manufacture helium bonded fuel elements. The fabrication of this fuel is difficult as it requires handling of pyrophoric carbide material in inert gas atmosphere and the number of process steps are more in comparison to oxide fuel fabrication [3, 4]. The fabrication flowsheet is shown in Fig. 1. The performance of the FBTR fuel was assessed at various stages of burnup in a comprehensive manner in an alpha-tight hot cell facility. Visual, dimensional and metallurgical examination of the fuel at different burnups, combined with modelling, was instrumental in taking the fuel to a record burnup of 165 000 MW·d/Te without any fuel pin failure in the core. This achievement has been possible through a combination of stringent fuel specifications, quality control during fabrication and inputs obtained from the detailed pre- and post-irradiation examinations of fuel at different stages, combined with the modelling of the behaviour of the fuel, clad and wrapper materials. Of late, the fissile material (Pu) recovered from reprocessing of carbide fuel has now been used for fabrication of fresh mixed carbide reload fuel. Closing the carbide fuel cycle has thus become complete, which is an important milestone in India's fast reactor fuel cycle.

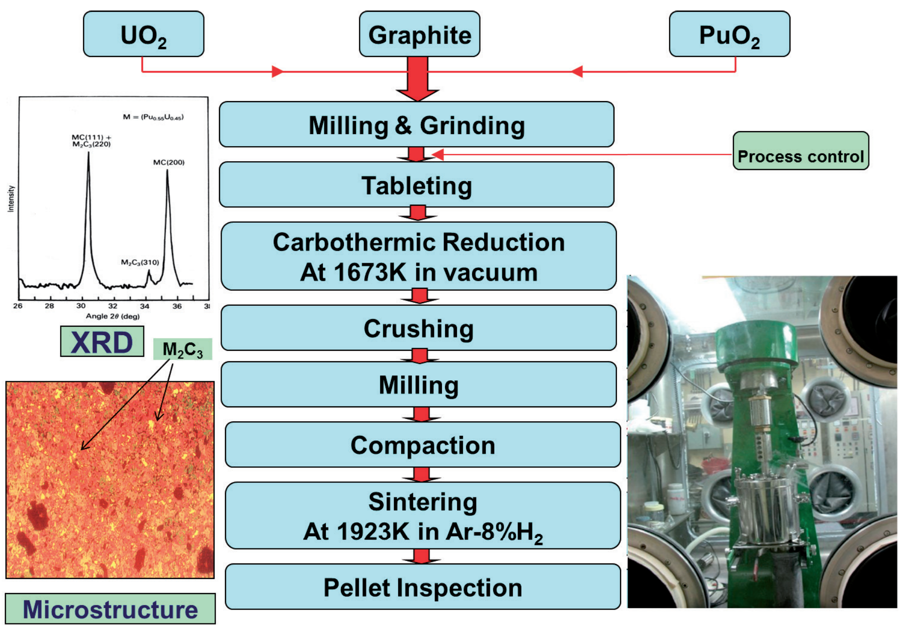

FIG. 1. Flowsheet for the fabrication of (U,Pu)C fuel for the FBTR.

At present, the FBTR operates with a mixed core consisting of mixed carbide and UO_2-45%PuO_2 MOX fuel. About 20% of the core has now been loaded with (U-45%Pu) MOX. Use of UO_2-45%PuO_2 as partial core of the FBTR strengthens the technology base required for fabrication of PFBR MOX fuel as it will be carried out in the same or similar fabrication line at the Advanced Fuel Fabrication Facility, Tarapur. As this fuel has higher oxygen potential and lower thermal conductivity, its performance will be a more conservative representation of PFBR fuel. The thermal conductivities of UO_2-45%PuO_2, along with those of mixed carbide, MK-I and MK-II compositions, are shown in Fig. 2. It indicates that for carbide fuel, thermal conductivity increases with temperature and the thermal conductivity of MK-I fuel is lower than that of MK-II up to about 1100 K [5]. The oxide has much lower conductivity than carbide, which decreases with increase in temperature. The fuel coolant (using high purity sodium) compatibility studies carried out at RMD have shown that UO_2-45%PuO_2 has acceptable compatibility with sodium. The studies have also indicated that at O/M = 2.00 the fuel has single phase and is stable under thermal cycling conditions [6].

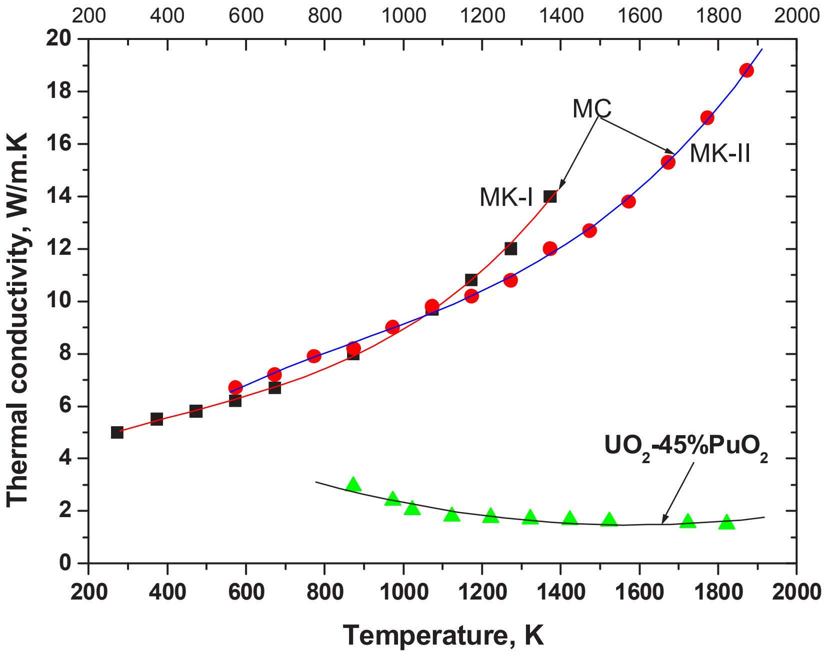

FIG. 2. Thermal conductivity of mixed carbide of composition MK-I ((Pu$_{0.70}$U$_{0.30}$)C), MK-II ((Pu$_{0.55}$U$_{0.45}$)C) and MOX ((U$_{0.45}$Pu$_{0.55}$)O$_2$) fuels as a function of temperature.

3. THE PFBR

Construction of the 500 MW(e) PFBR has been undertaken by the public sector company BHAVINI. MOX (U-Pu) fuel was selected as the driver fuel for the PFBR-500 because of the good technology base for (U,Pu) MOX fuel manufacture and oxide fuel reprocessing.

The PFBR core consists of 85 fuel assemblies of 21% Pu MOX and 96 fuel assemblies of 28% Pu MOX. The cladding material chosen for PFBR is D-9 (Ti modified SS 316; 20% cold worked). The fuel pin consists of annular MOX pellets of about 5.55 mm in diameter. The Advanced Fuel Fabrication Facility at BARC has taken up the fabrication of the MOX fuel pins for the first core of the PFBR. The critical technologies for production of homogeneous MOX fuel such as attritor milling, annular pellet production using rotary presses, dry centreless grinding (if necessary) and welding of D-9 using pulse TIG techniques have all been developed. The plant uses high level of automation for its operations with enhanced safety features for its glovebox lines [2]. The fabrication flowsheet for the manufacture of the PFBR fuel pin is given in Fig. 3. Typical annular MOX fuel pellets are shown in Fig. 4. The core structural components, including D-9 clad tubes, hex can, radial blanket and steel shielding assemblies, are sourced from Nuclear Fuel Complex, Hyderabad. Thermophysical properties such as thermal expansion coefficient, thermal conductivity, phase stability, hot hardness and fuel–clad–coolant compatibility studies for MOX fuels for the PFBR have been completed. These studies, along with fabrication technology developed, enhanced confidence in the use of MOX fuel in the PFBR.

3.1. Fabrication of MOX PFBR test fuel assembly

In view of the interest in (U-Pu) MOX for the PFBR, the FBTR has been used as a test bed for experimental irradiation of MOX fuel with specifications close to those of the PFBR fuel. For this purpose, a short length PFBR experimental MOX fuel assembly, with fuel having a composition of UO_2-29%PuO_2, was fabricated. In order to simulate the high linear heat rating of the PFBR, $U^{233}O_2$ was also used in making the fuel, along with natural UO_2 and PuO_2 . The 37 pin PFBR fuel assembly loaded in the centre of the FBTR core was irradiated at a linear power rating of 450 W/cm. The MOX pellets had the same annular geometry and density as for the PFBR MOX fuel pellets and were clad in D-9 tubes of the same design specification as those for the PFBR clad, but shorter in length. This fuel has already exceeded the design burnup of 100 GW·d/Te and has since been discharged. This has given confidence in the fuel design and manufacturing practices employed. PIE on this fuel has been done.

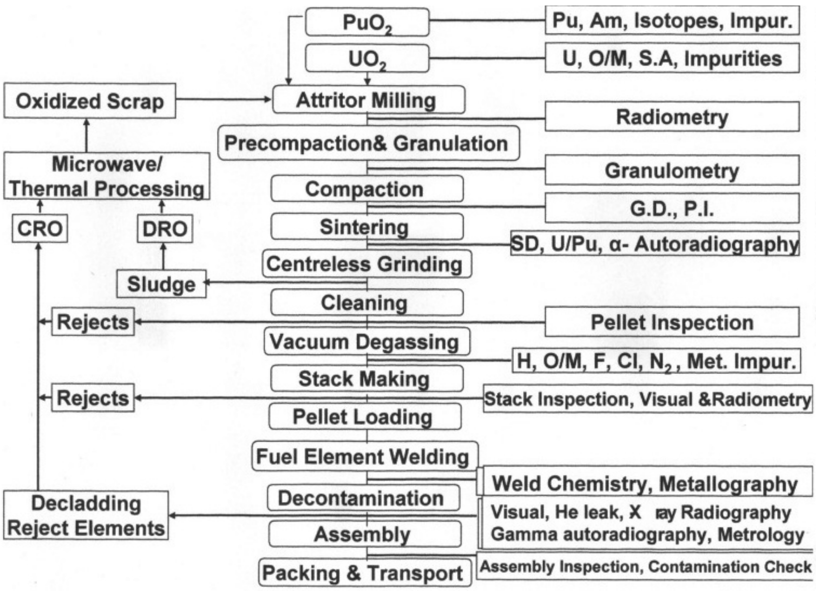

FIG. 3. Fabrication flowsheet for PFBR fuel.

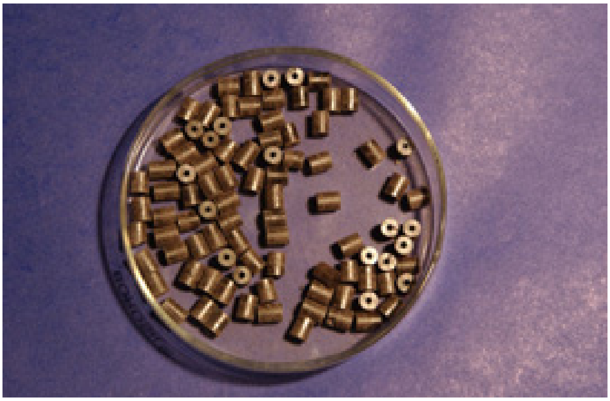

FIG. 4. Annular MOX pellet fabricated for the PFBR.

4. FAST REACTOR FUEL CYCLE FACILITY

Although the first few cores of MOX fuel for the PFBR will be manufactured at the Advanced Fuel Fabrication Facility (BARC), a co-located (with the PFBR) integrated Fast Reactor Fuel Cycle Facility is going to be set up at Kalpakkam. This facility will have fuel manufacturing, reprocessing and waste management plants to take care of the reload requirements for the PFBR. A high degree of automation and remote operation is being introduced to handle high burnup and multicycled Pu. The detailed engineering of this facility has been completed and the construction will be taken up shortly. Sol-gel based sphere-pac and SGMP techniques are also being investigated for possible application in future facilities.

5. DEVELOPMENT OF FUEL CYCLE FOR METALLIC FUELS FOR FAST BREEDER REACTORS

For the rapid growth of the fast reactor programme in India, it is essential to shift to the use of metal fuels in fast breeder reactors, which give a higher breeding ratio and lower doubling time. The use of metallic fuel along with the pyroprocess recycling will be less costly and proliferation resistant than oxide fuel reprocessing. The higher breeding ratios and thus shorter fuel doubling times for metal fuels arises due to the harder spectrum as compared to that of ceramic fuels. For a 500 MW(e) reactor, the oxide fuel could result in a breeding ratio of 1.09 and fuel doubling time of 40 years. With carbide fuel, the breeding ratio can be improved to 1.19 and the doubling time reduced to 20 years. Quantum increase in the breeding ratio is achieved with metallic fuels. U-Pu-Zr fuels with varying Zr contents have been studied [1]. It has been found that the breeding ratio increases with reduced Zr content. Thus, for a U-Pu binary fuel with 150 μm Zr liner on inner clad, the breeding ratio is calculated to be 1.56 with a fuel doubling time of 7 years. Table 1 gives the breeding ratios and doubling times for various fuels used in fast reactors [1].

Primarily, two design concepts have been proposed for the metallic fuel development programme for fast breeder reactors in India [6]. Two fuel concepts being explored are:

(i) Mechanically bonded pin with U-15wt%Pu alloy as fuel;
(ii) Sodium bonded pin with U-15wt% Pu-6wt% Zr alloy as fuel.

TABLE 1. BREEDING RATIO AND DOUBLING TIME FOR DIFFERENT TYPES OF FAST REACTOR FUEL

Fuel material	U-Pu oxide	U-Pu carbide	U-Pu-10% Zr	U-Pu-6% Zr	U-Pu with 150 µm Zr liner
Breeding ratio	1.09	1.19	1.36	1.47	1.56
Doubling time (y)	40	20	9.4	7.2	7

Figure 5 shows the cross-sections of a conventional sodium bonded fuel pin and a mechanically bonded fuel pin. Sodium acts as a thermal bond between the fuel (U-Pu-Zr) and cladding material in the sodium bonded fuel pin. In the mechanically bonded fuel pin, Zr is used as a barrier layer between the fuel and clad. The cladding with a Zr barrier layer will be swaged on the fuel slug and it is expected that there will not be any physical gap between barrier layer and fuel slug. Semicircular grooves placed diametrically opposite are provided in the fuel slug for the accommodation of irradiation induced fuel swelling. Helium gas is used to fill the groove region. The smear density, which is a key parameter for accommodation of fuel swelling, varies between 70% and 85% for mechanical bonded fuel and 70% for sodium bonded fuel [7–10].

Mechanically bonded fuel has the following advantage over the sodium bonded fuel:

- The fission gas can be located at the colder bottom of the fuel pin thus requiring lesser space for fission gas and hence giving a shorter pin length and more space for fuel/axial blanket.
- No issues of handling highly contaminated sodium in reprocessing of the burnt fuel and waste management.

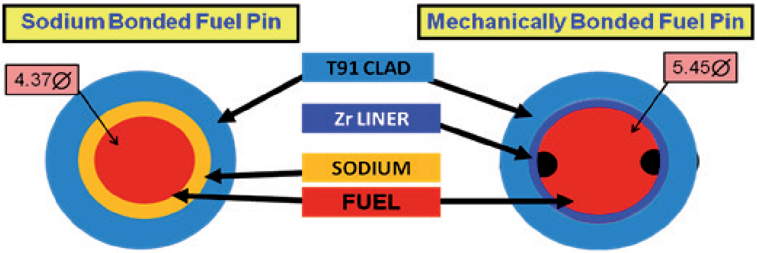

FIG. 5. Design concepts in metallic fuels.

- Owing to lower liquidus temperature of U-Pu than U-Pu-Zr, the fabrication temperature is lower, resulting in less Pu loss.

5.1. U-Zr system

U-6wt%Zr is a subsystem of U-15wt%Pu-6wt%Zr alloy, and is also proposed as a blanket material for the sodium bonded ternary U-Pu-Zr fuel. For the fabrication of metallic fuels, a demonstration facility has been set up in the Atomic Fuels Division, BARC, for injection casting of uranium rods in quartz moulds. This is followed by demoulding and end shearing. An automated system for inspection of fuel rods with respect to their mass, length, diameter, diameter variation along the length and internal and external porosities/voids has been a part of the fabrication flowsheet. The entire fabrication flowsheet for the fabrication of mechanically bonded U-Zr alloy is shown in Fig. 6. The facility has been designed for subsequent use in the fabrication and inspection of Pu-bearing metallic fuels. This facility has successfully been used to produce 5.0–6.0 mm diameter fuel slugs of varying lengths with random grain orientation which otherwise would require a number of thermomechanical treatments.

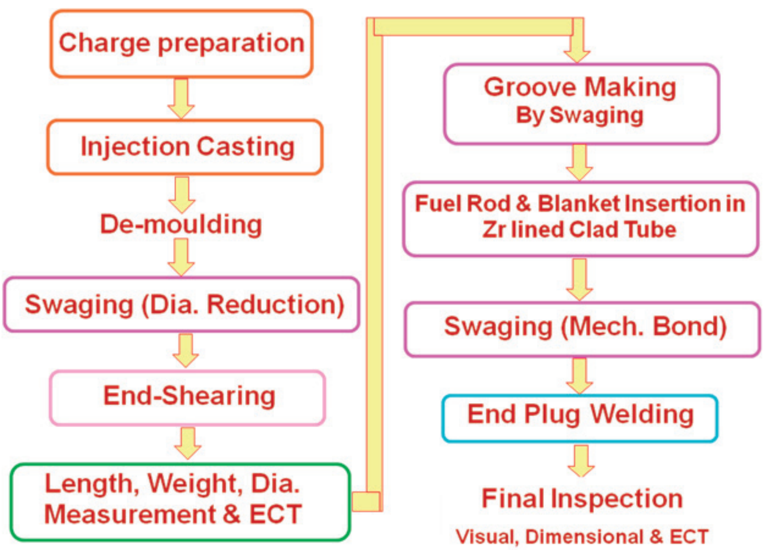

FIG. 6. Fabrication flowsheet for mechanically bonded U-Zr alloy.

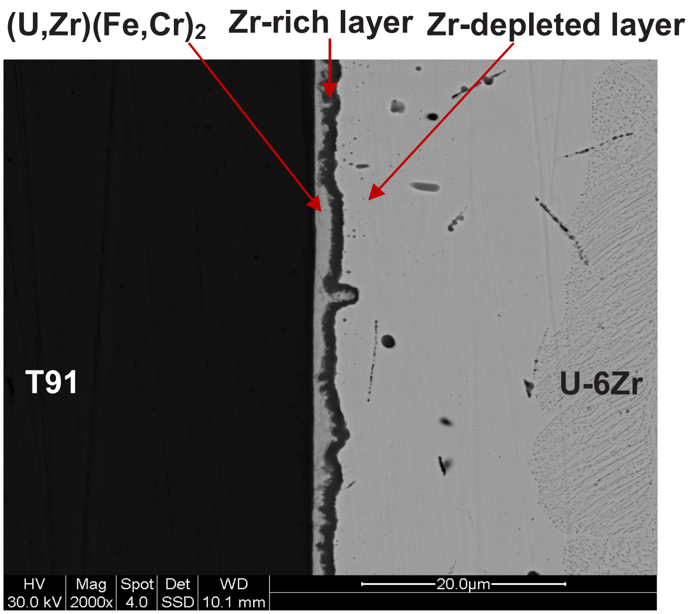

FIG. 7. The microstructure of the interdiffusion layer formed at the interface of U-6Zr/T91 couple after annealing at 973 K for 500 h.

The microstructural and high temperature behaviour of U-6wt%Zr alloy has been investigated at BARC recently. Thermophysical properties such as coefficient of thermal expansion, specific heat, thermal conductivity of the above alloy were determined. The hot hardness data of the U-6wt%Zr alloy was also generated from room temperature to 973 K. Apart from that, the fuel–clad chemical compatibility with T91 grade steel (low carbon, 9Cr-1Mo ferritic martensitic steel containing small amounts of V, Nb, Si, Mn) was also studied by diffusion couple experiment. The eutectic reaction temperature between U-6Zr alloy and the T91 steel system is an important data for the fuel designer and was found to be almost equal to that of U-Fe, i.e. 995 K. The diffusion couple experiment has shown that interdiffusion between U-6Zr and T91 at 973 K for 500 h results in the formation of a $(U,Zr)(Fe,Cr)_2$ type layer on the clad side and a Zr depleted layer on the fuel side. A Zr rich layer was found between these two which acts a fuel clad interdiffusion barrier (Fig. 7). At 1023 K, the U-6Zr/T91 couple reacted completely with each other, causing a complete meltdown of the clad in 100 h, and showed eutectic melted microstructures of U_6Fe, $U(Fe,Cr)_2$ and $Zr(Fe,Cr)_2$ phases [7].

5.2. U-15%Pu binary alloys

The mechanically bonded binary U-15wt%Pu fuel is a relatively new concept with very little international experience. No detailed studies have been reported in the open literature on U-15%Pu alloy. Hence, studies related to development of fast reactor fuels based on binary U-15%Pu alloy has been initiated in India for building a database on thermophysical and thermodynamic properties, fuel–clad compatibility, etc., which are essential to the fuel designer to optimize the design feature and to predict the in-reactor fuel behaviour. A detailed study on thermophysical properties of U-15%Pu alloy at high temperatures has been conducted and the following conclusions have been drawn [8, 9]:

- The XRD and microstructure of as-cast U-15%Pu alloy showed the presence of only an α phase.
- The solidus temperature of U-15Pu is 1248 K.
- The average coefficient of thermal expansion has been determined and found to be $18.58 \times 10^{-6} K^{-1}$ in the temperature range 300–823 K.
- The hardness showed the γ phase region of U−15%Pu is very soft.
- The eutectic reaction temperature between U-15%Pu alloy and T91 steel is 948 K, as shown in Fig. 8.
- The results of the U-15%Pu/Zr/T91 diffusion couple indicate that the Zr liner was effective in preventing fuel–clad chemical interaction at 973 K for 500 h. This is clearly shown in Fig. 9.

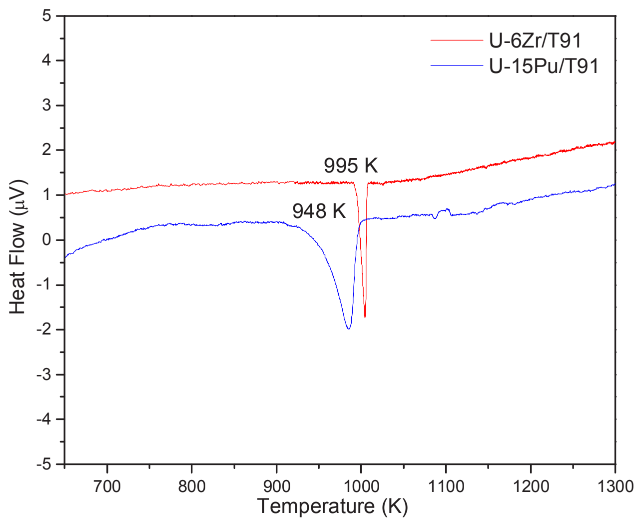

FIG. 8. Eutectic temperature between U-15%Pu fuel and T91 cladding. For comparison, the U-6Zr/T91 eutectic temperature is also shown.

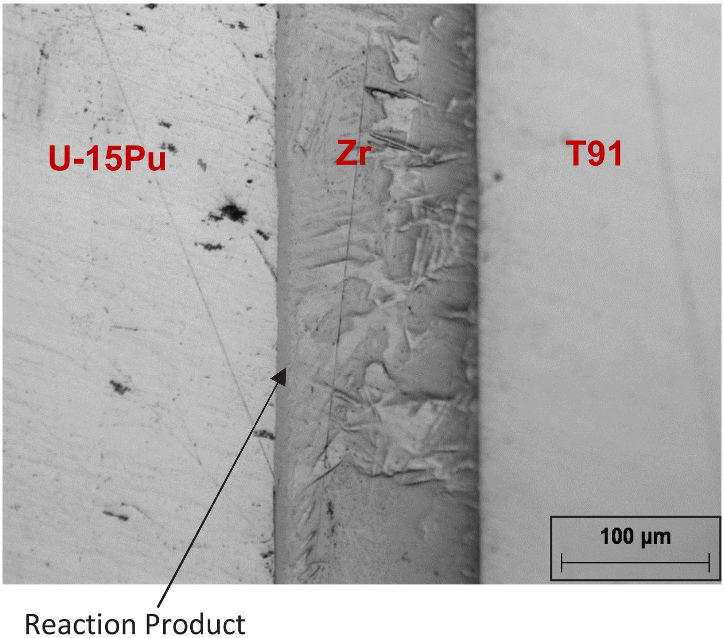

FIG. 9. Effectiveness of Zr barrier in preventing the reaction between U-15%Pu alloy and T91 cladding after heating the couple for 500 h at 973K.

6. CERMET FUELS

CERMET nuclear fuels have a significant potential to enhance fuel performance because of low internal fuel temperatures and low stored energy. The combination of these benefits with high burnup capability and favourable neutronic properties may make them very attractive in advanced nuclear fuel cycles. CERMET fuels consist of ceramic fuel particles such as UO_2 or PuO_2 dispersed within a metal matrix of U or U-Mo alloy. The high thermal conductivity of the metal matrix leads to a cold fuel pin, resulting in low central temperature, low stored energy and low thermal gradient. High fission gas retention in CERMET fuel is expected owing not only to the low operating temperature but also to the fact that the metal constitutes an additional non-porous barrier. This improvement in terms of fission gas retention makes it possible to increase the burnup in a very significant manner. Another likely advantage of CERMET fuel elements over conventional metallic and ceramic fuels is their irradiation stability. This is mainly due to the fact that the fission products are retained near to the dispersed fuel particles [1, 2].

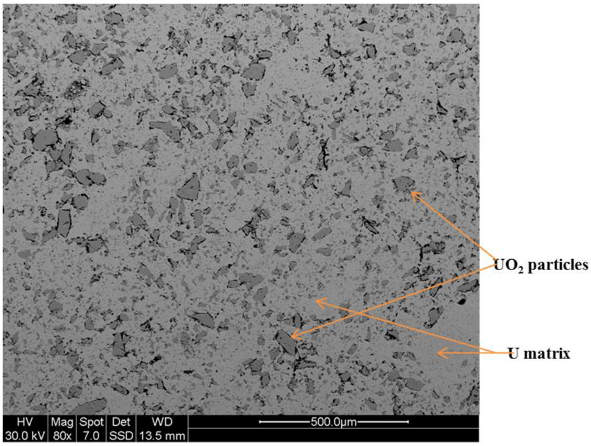

FIG. 10. Microstructure of U-15%UO_2 CERMET fuel.

In order to combine the benefits of both oxide and metallic fuels, which would lead to obtaining optimum fuel properties, work has been initiated at the BARC on U based CERMET fuels for fast reactors which has the potential for achieving a higher breeding ratio. U-15%UO_2 and U-30%UO_2 CERMET fuel pellets were prepared on an experimental scale by the powder metallurgy route. The compatibility studies of the above with T91 cladding and the evaluation of other thermophysical properties are in progress. This work will be extended in the future to the development of U-PuO_2 CERMET fuels. A typical microstructure of a U-15%UO_2 CERMET pellet is shown in Fig. 10.

7. CONCLUSIONS

The Indian nuclear power programme is based on a closed nuclear fuel cycle for efficient utilization of its nuclear resources. Development of mixed carbide for the FBTR and MOX fuel for the PFBR has provided the confidence required to manufacture fuels for fast reactors. The relatively low fissile density of MOX fuel made the higher fissile density fuels, such as metal, mixed carbide and mixed nitride, attractive for better breeding performance. Metallic fuel is reported to be very efficient from the point of view of high breeding ratio and low doubling time. Metallic fuel, in combination with pyro-metallurgical reprocessing and injection casting, is very promising with regard to the integrated fast reactor with co-location of reactor, fuel fabrication and reprocessing facilities. The sustainability of nuclear energy in India will depend heavily on the development and deployment of high breeding fuels, i.e. metal or CERMET for fast reactors.

The challenges ahead include development of remote and automated fuel reprocessing/refabrication technology and detailed analysis of factors affecting the fuel cycle cost.

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RECENT ADVANCES IN FUEL FOR FAST REACTORS: SYNTHESIS, PROPERTIES, SAFETY PERFORMANCE

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Abstract

This presentation provides an overview of advances made in fast reactor fuel safety research in the last decades. MOX fuels were selected for many fast reactors. The need to minimize waste through partitioning and transmutation strategies has seen new fuel forms evolve, including fertile and non-fertile targets to host the minor actinides. Despite the proven safety performance of MOX fuels and the progress made in nitride and carbide driver fuels and also in minor actinide oxide fuel research programmes, improvements in knowledge and understanding of the safety performance of these fuels can be made. Above all, breakthroughs in simulation and modelling need to be harnessed for dedicated experiments, leading to even more reliable and robust engineering codes for the qualification of all fast reactor fuels.

1. INTRODUCTION

Fuels for fast reactors operate under extreme conditions, owing to the high power density of the reactor core. They have to withstand damage due to neutrons and large thermal gradients. They must show an adequate pellet cladding mechanical interaction and benign pellet cladding chemical interaction. They must incorporate fission products and accommodate fission gases either in the crystal lattice, in bubbles in the fuel, or permit their release to the plenum. If minor actinides are present in the fuel, as is foreseen in partitioning and transmutation strategies, then helium must be accommodated too.

The chemical form of the fuel has largely concentrated on oxides, especially in Europe. Metal fuels have been favoured in the United States of America [1] and in other countries owing to their faster doubling time. Owing to their high swelling rates, however, they are usually operated with a wide pellet to clad gap, whose thermal conductivity is dramatically improved by the introduction of a sodium bond. Despite the high conductivity of these fuels, their operating temperature, like oxides, is about 80% of their melting point. Nitrides and carbides possess much higher melting points than metal fuel, and also have higher thermal conductivity than the oxides (see Table 1), which when coupled to their high metal content provide strong reasons to advocate their deployment in fast reactors.

TABLE 1. REPRESENTATIVE PHYSICAL PROPERTIES OF (U,20%Pu) OXIDE, CARBIDE AND NITRIDE FUELS

Fuel type	$(U,Pu)O_2$	(U,Pu)C	(U,Pu)N
Melting point (°C)	2750	2480	2650
Boiling point (°C)	3150	4280	n/a
Theoretical density ($g\cdot cm^{-3}$)	11.06	13.6	14.3
Heavy metal density ($g\cdot cm^{-3}$)	9.7	12.9	13.5
Thermal conductivity ($W\cdot m^{-1}K^{-1}$ at 1000°C)	3.0	20.5	20.0
Linear coefficient of thermal expansion ($\times 10^{-6}$)	10.7	14.7	12.5

Historically, in Europe at least, oxide fuels were selected for fast reactor applications, despite appreciable parallel efforts on both carbide and nitride fuels. The minimization of nuclear fuel cycle waste became an important research area in the last 20 years, whereby the long lived radioisotopes (mostly the minor actinides neptunium, americium and curium) should be partitioned from the nuclear waste for recycling and transmutation in fast reactors [2]. Reactors operating with fast neutrons are the most efficient transmutation devices. They can be conventional in form or non-conventional (i.e. the accelerator driven system). Early minor actinide fuel safety tests (SUPERFACT) were performed by the CEA and JRC-ITU in the Phenix reactor, and no detrimental behaviours were found for fuels representative of the so-called homogeneous and heterogeneous minor actinide recycling concepts. In the former strategy, 1–3% of minor actinides are added to all the conventional mixed uranium-plutonium fuel in the reactor core. In the latter strategy, one considers dedicated targets, where the majority of the core remains standard, but the periphery accommodates dedicated assemblies with high minor actinide quantities in a matrix with no plutonium in the fresh fuel, e.g. $(U,MA)O_2$. Inert matrices, to support either non-fissile or fissile fuel particles, have the advantage that they produce no further Pu or minor actinides during irradiation, and thereby optimize the transmutation efficiency. Of course, there is far less knowledge on the safety of such inert matrix fuels.

This paper outlines some of the progress made in the last decade and concludes with some general recommendations for the future.

2. FUEL SYNTHESIS

2.1. Conventional oxide fuels

In Europe, the operated fast reactors were mainly fuelled by mixed uranium-plutonium oxides. Though there were forays into exotic synthetic methods based on liquid processing, the main synthesis routes deployed were based on powder metallurgy. This remains the same today. Stored UO_2 and PuO_2 powders are mixed in the appropriate proportions. Without milling, the final product invariably exhibits two ceramic phases (CERCER) consisting of islands of PuO_2 distributed in a UO_2 matrix. Ball milling the precursor powders can result in a near perfect distribution of the Pu in the mixed oxide product.

Traditional fuel pellets are formed by the compaction of the powders in bi-directional uniaxial presses followed by sintering of the pellets in the desired atmosphere to control the final oxygen to metal (O/M) ratio. The latter value is normally chosen to be somewhat less than 2.00 to minimize risk of oxidation of the stainless steel cladding on contact with the fuel during irradiation.

Major breakthroughs in mixed oxide (MOX) fuel synthesis have not been reported in recent years, except for developments in Japan [3], where the starting powder is a 50:50 mixture of uranium-plutonium oxide, generated by a microwave conversion process.

Improvements in the safety of the synthesis have been investigated at the JRC-ITU laboratory in an attempt to avoid the milling step, which produces very fine powders [4]. In this route, an additive (bentonite) is added to the UO_2-PuO_2 powder blend without milling. This additive acts as a liquid phase sintering aid, and greatly enhances the diffusion between the UO_2 and PuO_2 particles, resulting in a near homogeneous solid solution. A potential caveat lies in the slightly reduced density of the fuel pellets.

2.2. Nitride and carbide fuels

The synthesis of nitride and carbide fuels is more demanding than the oxide counterparts. Owing to their sensitivity to oxygen, a highly pure atmosphere is required in the gloveboxes. Only in India have carbide fuels been produced on a large scale, though programmes in many other countries advocate these fuel forms. The carbothermal reduction route remains the synthesis route of choice. It is a complex approach and requires a parametric investigation to optimize the synthesis. Excess carbon, beyond the stoichiometric quantity, is required to achieve complete reaction. The excess is removed in the final step by treatment in hydrogen.

Research groups at BARC in India, ORNL in the USA, PSI in Switzerland and JRC-ITU in Germany have investigated methods to diminish the problems of pyrophoricity by synthesizing the precursor UO_2-PuO_2-C mixture in the form of large (>50 μm) particles. This has been achieved in a sol-gel step using external or internal gelation. Such methods have the advantage that the actinide precursor oxide for the carbothermal reduction is already in the form of a solid solution — $(U,Pu)O_2$.

During the carbothermal reduction, the powders or particles sinter to such an extent that they are unsuitable for pelletization. Again, high energy milling is required to improve the synthesis. In 2009, Muta et al. reported for the first time the use of spark plasma sintering for the production of high quality, high density nitride pellets, eliminating the undesired milling step [5]. The pressed compact is mounted in a graphite die through which a pulsed current is passed. The sample and die increase in temperature, while a pressure is held on the pellet. Typical cycle times for a single pellet are of the order of 20 minutes, with only five minutes at high temperature, which is an advantage in minimizing losses due to actinide vaporization.

2.3. Minor actinide bearing fuels

Minor actinide bearing fuels can also be synthesized by classical powder metallurgical routes, as has been achieved at the JAEA and CEA laboratories in Japan and France, respectively. There are safety issues related with powder handling, especially on large scales. Dusts migrate into many regions of the facility, ultimately making maintenance difficult.

Liquid conversion routes for minor actinides have been developed at the CEA and JRC-ITU laboratories [6, 7]. The former concentrates on the preparation of the precursor powder using an oxalate precipitation similar to that used today to precipitate Pu oxalate from nitric acid solutions at reprocessing plants. Excellent results have been obtained, especially in view of solid solutions and in terms of the pellet produced (density, visual appearance, etc.). Nevertheless, dust remains an issue.

Sol-gel routes based on external and internal gelation can overcome the issue of dust. The advantages of these methods, though often acclaimed in the past, were never brought to fruition on a large scale. There is potential for these methods, at least at the laboratory scale, to synthesize high quality materials for basic property measurements. The JRC-ITU has taken this procedure one step further and uses sol-gel methods to prepare well defined particles with diameters in excess of 50 μm. Their porosity opens the path for a further synthesis innovation, wherein an americium nitrate solution is infiltrated into the beads. Thermal treatment to evaporate water and convert the americium nitrate to oxide provides a powder ready for compaction. Owing to the resounding sintering nature of the sol-gel material and the very small particle size of the AmO_2, high quality solid solutions are obtained.

There have been very few attempts to produce minor actinide bearing nitride or carbide fuels. The volatility of Am is particularly problematic. Table 2 presents relevant safety data from the synthesis of carbide and nitride fuels for the NIMPHE fuel safety irradiation test. The Am/(Pu+Am) increased for the nitride fuel and is simply due to the decay of ^{241}Pu to ^{241}Am. In contrast to the nitride, however, a significant decrease (about 70%) of the Am/(Pu+Am) ratio occurs during the synthesis of the carbide fuel and is due to Am vaporization during the synthesis.

TABLE 2. FUEL SYNTHESIS DATA FROM THE NIMPHE IRRADIATION TEST

	^{239}Pu	^{241}Pu	Am/(Pu+Am)
As delivered	74.6	2.62	0.365
Nitride product	74.7	2.573	0.436
Carbide product	74.7	2.58	0.112

3. FUEL PROPERTIES

The properties of MOX fuels were thought to be well known. Recent measurments at the JRC-ITU on the melting point of PuO_2, however, have shown that it is 300 K higher than previously accepted [8]. In the composition range for nuclear fuels (~20–30% Pu), there is little change from the previously measured values, but the models used in the thermodynamic modelling of such solid solutions need to take account of the higher melting point of PuO_2.

The JAEA has been been particularly active in the determination of the properties of fresh minor actinide bearing MOX fuels. A major focus has been the oxygen potential and the thermal conductivity, which tend to degrade with addition of minor actinides [9].

Data on carbide and nitride fuels are not nearly as far reaching as those available on oxide fuels. Furthermore, property measurements have in the main concentrated on fresh fuel, with studies on irradiated fuel being far fewer.

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 experiment. The F-Bridge project [10], supported by the European Commission in Europe is a case in point. Over 20 research institutes came together in a project designed to bridge not just experiment and theory, but also (and for the future decidely important) length and time scales, i.e. from the nanometre to the metre, from the picosecond to the year.

4. FUEL SAFETY PERFORMANCE

MOX fuels have been operated in a number of fast reactors (e.g. Phenix, Superphenix, DFR, Monju, Joyo, BOR-60) and in Europe at least are the primary choice for ongoing fast reactor projects now in their conceptual phases, whether sodium (ASTRID), lead (ALFRED), lead–bismuth (MYRRHA) or helium (ALLEGRO) cooled. Particularly for sodium cooled fast reactors, MOX fuel is at an advanced level of maturity. Once compatibility with Pb and Pb–Bi under normal and off normal conditions can be ascertained, much of the operational knowledge achieved for sodium cooled reactors can be used in the safety evaluation of the other heavy metal cooled fast reactors.

The gas cooled fast reactor is a rather special case. The ALLEGRO reactor should be fuelled with a first core consisting of MOX fuel in a stainless steel cladding. Positions should be available therein for the testing of the truly foreseen fuel, a mixed carbide (or nitride) encapsulated in a SiC-SiC_f reinforced cladding, permitting the development steps for the second fully ceramic ALLEGRO core. Much development, safety testing and assessment work is still needed.

The irradiation performance of minor actinide fuels has been investigated in fast reactors in the last decade. An exciting result was found in the AM1 test [11], inspired and performed by the JAEA in Joyo. Several minor actinide bearing MOX fuels, consistent with homogeneous recycling of minor actinides in fast reactors, were irradiated for 10 minutes and for 24 hours. Even after 10 minutes, the onset of the central hole formation could be observed. After 24 hours, the formation of the central hole was fully complete.

A number of irradiation tests on minor actinide bearing fuels have been performed in the Phenix reactor. The majority remain to be analysed in depth. The ECRIX experiment, performed by the CEA [12], investigated the irradiation behaviour of $AmO_{1.6}$ dispersed in an MgO matrix. The fuel pin inside the capsule was surrounded by neutron moderators to tune the neutron spectrum for increased transmutation. Both B_4C (ECRIX-B) and ZrH_2 (ECRIX-H) were tested. No detrimental performance behaviour was found.

The FUTURIX irradiation test was prepared and run jointly by the CEA, US DOE and JRC-ITU. Candidate minor actinide bearing metal and nitride fuels were prepared by the INL and LANL, respectively, while CERCER and CERMET fuels were prepared by the CEA and JRC-ITU. In the CEA and JRC-ITU samples, the fissile phase was $(Pu,Am)O_{2-x}$, while the diluant was MgO and Mo, respectively. These CERCER and CERMET fuels are foreseen as candidate fuels for an accelerator driven system. The irradiated fuel pins remain at Phenix awaiting shipment to various hot cell laboratories for PIE.

Minor actinides can also be recycled in fast reactors in a fully heterogeneous mode, i.e. in regions close to the periphery of the reactor core. Though much effort has been given to inert matrices (e.g. MgO), the focus today is on $(U,MA)O_{2-x}$. A key safety related parameter for such fuels is the helium behaviour, as it is produced in large amounts in such transmutation targets and must be accommodated. Furthermore, safety issues during reactor start-up and during transients must also be examined.

5. CONCLUSIONS AND OUTLOOK

Fast reactor MOX fuel with and without minor actinide (homogeneous minor actinide recycle concept) exhibits major restructuring (formation of a central hole, columnar grains, etc.). The mechanisms involved are not well tested nor proven. This could be achieved by determination of the local O/M ratio, which must influence uranium displacement down the temperature gradient. 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.

Spent fuel from existing and closed fast reactors represents an important knowledge legacy, opening up the possibility for dedicated studies using techniques not available 40 years ago (e.g. thermal conductivity, isotope radial distribution, vaporization behaviour), when these fuels were originally licensed.

The behaviour of Am in fuel has been studied only to a very limited extent. Helium behaviour in unirradiated UO_2 needs further experimental verification in model and real systems. This is especially important for targets (heterogeneous minor actinide recycle) as the helium produced depends directly on the minor actinide content. Separate effect irradiation tests (e.g. disk isothermal tests) can yield mechanistic information. Novel designs for helium management (mitigation of swelling) need consideration and testing in integral irradiation tests. Thorough safety analyses of targets, operating at relatively low temperatures, need to be made in view of fuel power transients, which could cause massive He release.

Synthesis of carbide and nitride fuels would benefit from a major radical innovation to improve their reproducibility. Integration in the pyrometallurgy recycle concept (as proposed in Japan) using direct carbiding or nitriding of metal, or direct precipitation (of the nitride or carbide) from such melts is an option. New precursors could be designed to enable simpler and more reliable conversion from aqueous solution to the precursor powder, avoiding the oxide step. The establishment of properties of carbides and nitrides at high temperatures needs urgent attention to improve or at least recognize the boundary limits for their synthesis.

Modelling and simulation of fuel properties and underlying mechanisms has made major progress in the last decade. This will continue. To take maximum advantage, strong coupling to experiment is required. One can envisage improved use of modelling for better designed experiments, geared to 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.

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STATUS OF SFR METAL FUEL DEVELOPMENT

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Abstract

Metal fuel, U-TRU-Zr is being developed for the Generation-IV sodium cooled fast reactor. TRU recovered through the pyro-electrochemical processing of spent LWR fuel, which is highly radioactive owing to minor actinides and also contains the chemically active lanthanide elements, is used to fabricate the metal fuel. Therefore, a simple fuel slug casting system which can control the volatile elements and handle chemically active lanthanide elements is being developed. Ferritic martensitic steel cladding is being developed to be used for high burnup fuel. Barrier technology to prevent interaction between metal fuel and cladding was investigated and Cr electroplating on the inner surface of the cladding was developed. Fabricated metal fuel rods including Cr-plated barrier cladding were irradiated in the reactor and post-irradiation examination is currently ongoing.

1. INTRODUCTION

As the fuel for the Generation-IV sodium cooled fast reactor (SFR), metal fuel was selected to meet the targets of the Generation-IV reactor such as economy, safety, sustainability and proliferation resistance [1]. For sustainability and proliferation resistance, long lived minor actinides (MAs) such as Np, Am and Cm are transmuted in the reactor. To enhance economy, the fuel will be irradiated up to high burnup. Metal fuel has compatibility with sodium reactor coolant which guarantees flexibility and margin in reactor operation [2]. The higher thermal conductivity of metal fuel and adoption of fuel design with sodium fuel gap can keep fuel temperature low during irradiation. Therefore, an SFR using metal fuel can be operated with passive safety, which implies that fuel integrity is maintained during transients without support of an active reactor cooling system. Pyro-electrochemical processing of LWR spent fuels extracts uranium and TRU (transuranic, Pu and MA), and separates fission products for disposal. Then, recovered uranium and TRU are used to fabricate the metal fuel, U-Pu-MA-Zr. The technical challenges of U-Pu-MA-Zr fuel are fuel fabrication and irradiation performance up to high burnup.

2. METAL FUEL DEVELOPMENT

2.1. Fuel fabrication

Since metal fuel containing MAs is highly radioactive, U-Pu-MA-Zr fuel should be fabricated in the radiation shielded hot cell. Therefore, fuel fabrication technology with a high reliability, simplicity and easy maintenance is necessary.

In the TRU materials recovered through pyro-electrochemical processing, there are certain amounts (comparable to MAs) of impurities such as Nd, Ce, Pr and La (lanthanides), which have similar electrochemical characteristics to MAs. In the casting of metal fuel, U-Pu-Np-Am-Cm-Ln-Zr, vaporization of Am and interaction of chemically active Ln elements with the casting crucible are of concern. To supress vaporization of Am, pressurization of the casting chamber atmosphere is necessary.

Although injection casting has been a well-established fabrication method for metal fuel for decades [3], Am addition to the metal fuel hampers conventional fuel fabrication processes because of the high vapour pressure of Am at the melting temperature of uranium alloys [4]. A gravity fuel casting system, which can control transport of volatile elements during melting of a fuel alloy with MAs, has been developed. The melt in a crucible is cast into the mould under the crucible through a distributer by gravity, as shown in Fig. 1 [5]. The gravity casting system is more suitable for pressurization of the chamber during fuel casting than conventional injection casting. Volatile Mn was used to simulate volatile Am for fuel casting tests.

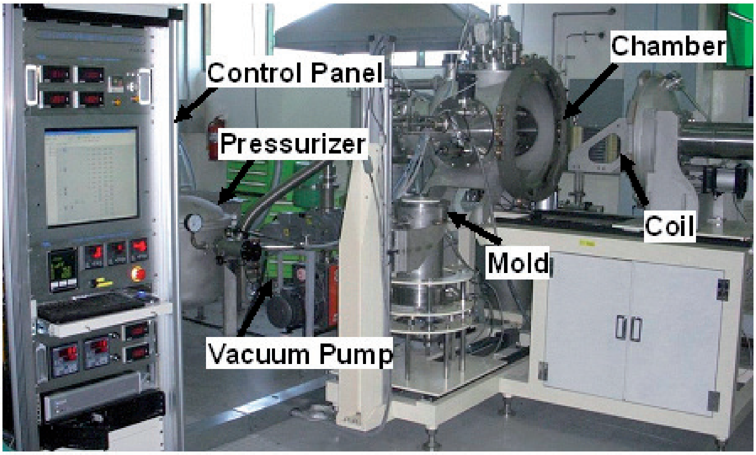


FIG. 1. Gravity casting system.

Optimization of the fuel casting process has been performed to obtain a sound fuel slug. By using the advanced fuel casting system, U-10wt%Zr and U-10wt%Zr-5wt%Ln(Ln: Nd 53wt%, Ce 25wt%, Pr 16wt%, La 16wt%) fuel slugs were fabricated, as shown in Fig. 2. Gamma radiography was performed to detect internal defects such as cracks and pores inside the metallic fuel slugs. The microstructures were examined and thermomechanical property tests, such as heat capacity, thermal expansion and high temperature tensile tests, conducted. The capability of the advanced fuel casting system to control volatile elements during casting is being investigated by using volatile manganese, and fuel slugs (U-10wt%Zr-5wt%Mn) were fabricated without significant loss of manganese.

The effects of lanthanide element additions on the characteristics of U-10wt%Zr alloy were investigated, as shown in Fig. 3. The disperse precipitates containing elemental Ce in a U-10wt%Zr-5wt%Ce alloy were homogeneously and finely distributed. Some precipitates, identified as Zr rich precipitates, were observed in the U-10wt%Zr-5wt%Ce fuel matrix. A laminar structure with a thickness of about 0.2 μm was observed in the matrix of the U-10wt%Zr-5wt%Ce fuel slugs. To mitigate or prevent interaction of lanthanides with the crucible, a surface coating is applied to the crucible. Plasma coated Y_2O_3 showed good performance as a coating material.

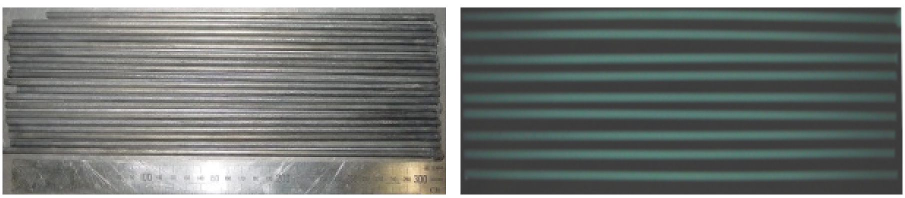


FIG. 2. U-10wt%Zr-5wt%Ln fuel slugs (5 mm diameter × 300 mm long) and gamma radiograph.

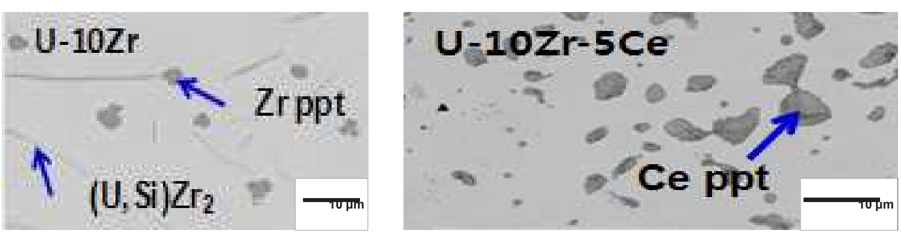


FIG. 3. Microstructure of the fuel slugs.

Particulate fuel fabrication is investigated as an innovative option. Metal fuel particles were fabricated by a centrifugal atomization process, as shown in Fig. 4. The size of fuel particles can be controlled from tens to hundreds of microns. Then, fuel particles will be vibro-packed into the cladding or consolidated through sintering. In this process, a quartz mould is not necessary for metal fuel fabrication. There is also the possibility that TRU particles can be directly fabricated by using the recovered TRU ingot from pyro-electrochemical processing and then mixing with particles of uranium and zirconium, followed by sintering. This process can alleviate the TRU ingot melting process after recovery by the pyro-electrochemical process.

The preliminary design of the metal fuel fabrication facility has been performed. Design requirements for operating and maintaining the facility were investigated and the safeguards concept was assessed. To demonstrate remote operability of fuel fabrication equipment, a mock-up test facility with glass windows and master-slave manipulators will be constructed for the next phase of research.

2.2. Cladding development

Among the fuel components of cladding, duct, wire, and top and bottom end pieces, in addition to the fuel slug, cladding is the most important in maintaining the integrity and safety of the fuel and the reactor since it prevents a release of radioactive fission products out of a fuel rod. It should withstand high temperature and high radiation damage conditions. Cladding with a high creep resistance at a high temperature, and with both low swelling and high ductility up to a high neutron fluence is necessary for the Generation-IV SFR fuel. Therefore, a high performance ferritic-martensitic steel (FMS) cladding is being developed through optimization of alloy compositions, microstructure and fabrication processes.

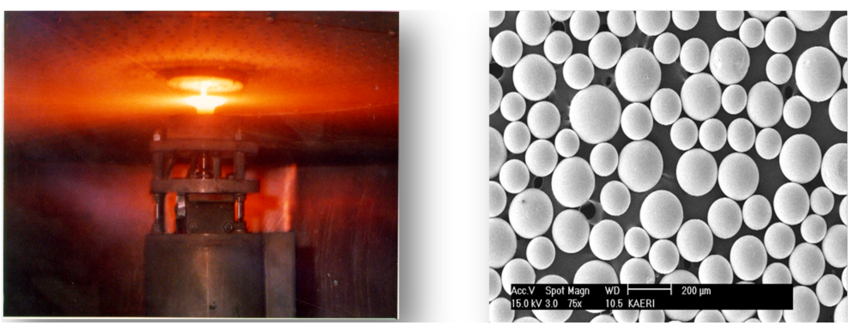


FIG. 4. U-10Zr fuel atomizing process and fuel particles.

FMS is the cladding material for the SFR metal fuel owing to its high thermal conductivity, low thermal expansion and excellent irradiation damage resistances [6]. When these steels are applied as the fuel cladding in the Generation-IV SFR, their maximum temperatures are expected to approach 650°C and the maximum irradiation damage by fast neutrons is expected to be higher than 200 dpa (displacement per atom). The fuel cladding should thus have good mechanical properties, such as tensile strength and creep resistance at high temperatures.

HT9 and Grade 92 FMS cladding tubes, OD 7.0 mm, T 0.6 mm, and L 3000 mm were fabricated and characterized. Cladding tube fabrication processes such as cold work (e.g. pilgering and drawing) and heat treatment were investigated. Mechanical properties and tensile properties and creep behaviour were tested and mictrostructure and dimensional variation and surface roughness were measured to verify meeting the specification requirements (see Fig. 5).

New FMS alloy is also being developed on the basis of Grade 92. Experimental FM steels were designed, focusing on optimization of minor alloying elements such as B, Nb, Ta and C for better mechanical properties at high temperature [7–9]. The creep rupture strength of new FMS alloys showed improvement of over 35% compared to the HT9, and better than Grade 92. The cladding tube of the new FMS alloy will be fabricated and irradiation tests will follow.

2.3. Fuel performance evaluation

Fuel design, fabrication technology and fuel components should be verified by performance tests and evaluations. Fuel irradiation behaviour under all the postulated irradiation conditions needs to be predicted through performance modelling. To resolve one of the technical issues in a metal fuel, the possibility of eutectic melting between fuel metal and cladding [10], a barrier between fuel metal and cladding, was investigated. To study interactions between the fuel slug and cladding, such as eutectic melting, diffusion couple tests of U-Zr-Ln together with FMS such as HT9 were carried out. Diffusion couple tests were also performed by inserting the barrier materials, such as Zr, Nb, Ti, Mo, Ta, V and Cr, between fuel slug and cladding. Among these barriers, V and Cr exhibited the most promising performance.

Another approach to block direct interaction between fuel metal slug and cladding is to build a barrier in the fuel slug. Installation of a thin and tight barrier on the surface of the fuel metal slug by forming fuel oxide, nitride or carbide is being investigated.

After scoping various coating methods, Cr electroplating has been selected as one of the probable candidates because it is cost effective and easily applicable to smaller tube geometry, compared to the other methods. However, it was revealed that when plating under conventional conditions, numerous cracks were generated during the plating which acted as the diffusion path for the fuel component during the diffusion couple test. Research has focused on reducing such cracking to enhance Cr barrier performance. Pulse plating was by altering current density with time and heat treatment after the plating was performed to reduce residual stress, which induces internal cracking [11]. Cr of 20 micron thickness has been uniformly plated at the inner surface of the 9Cr-2W FMS tube having a 4.6 mm inner diameter which was used for irradiation tests.

Fuel irradiation testing was performed in the HANARO research reactor under simulated fast reactor conditions such as temperature, fission density and sodium fuel gap bonding. Twelve fuel rodlets were inserted with varying fuel slug composition (U-10%Zr-(0, 6 Ce)). Ce was selected as a representative lanthanide element. Four fuel rods with Cr plated barrier claddings were also included. Thermal neutrons are partially shielded by surrounding the capsule with neutron absorbers such as Hf. He gap between fuel and cladding is filled with a bonding material (Na). The cladding is sealed with the outer tube. Cladding temperature is raised by introducing the He filled gap between the cladding and the sealed tube. Figure 6 shows a schematic diagram of the irradiation capsule and coolant channel cross-section [12]. Irradiation was conducted between November 2010 and January 2012, reaching a burnup of 2.7%.

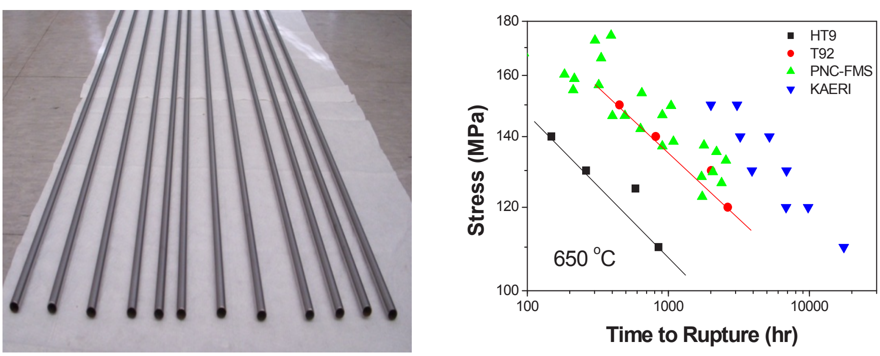


FIG. 5. HT9 cladding tube and creep test results of advanced FMS.

Post-irradiation examination of the irradiated fuels is currently ongoing. Representative destructive tests are intended to measure or observe fuel burnup, the microstructure, fission gas release and the constituent redistribution. Figure 7 shows the preliminary results of cut fuel rods. Transient behaviour tests under simulated conditions will be performed by using the irradiated fuel in the hot cell.

A new metal fuel performance code, called PUMA (Performance of Uranium Metal fuel rod Analysis code) is being developed. Multidimensional and multiphysical phenomena in nuclear fuels are treated as a set of monodimensional-coupled problems which encompass heat, displacement, fuel constituent redistribution and fission gas release. Rather than uncoupling these coupled equations as in conventional fuel performance codes, effort is being put into obtaining fully coupled solutions by relying on the recent advances in numerical analysis. Figure 8 shows a schematic diagram for a fully coupled approach. Coupling between temperature and fuel constituent was found to be made with relative ease by employing an ordinary differential equation solution [13]. The coupling between the mechanical equilibrium equation and a set of stiff kinetics equations for fission gas release is accomplished using a one-level Newton scheme by using backward differentiation formula. Displacement equations from a 1-D finite element formulation of the mechanical equilibrium equation are solved simultaneously with the stress equation, creep equation, swelling equation and FGR equations in the GRSIS model [14]. Figure 9 shows the variation of the number of bubbles and hydrostatic pressure over time. The two variables are interrelated. With a little effort, this methodology can be extended to model fuel–clad mechanical interaction and to attach additional physics factors such as a thermal equation and a chemical diffusion equation for fuel constituent redistribution modelling.

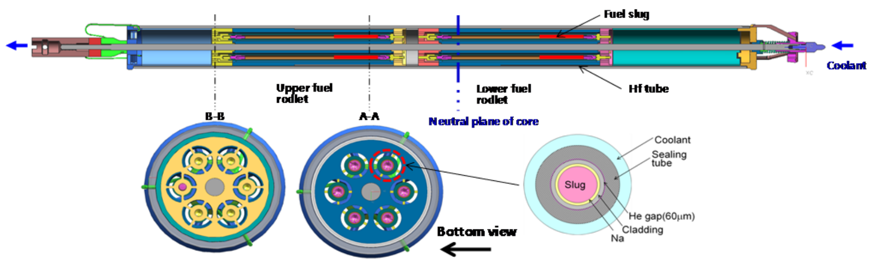
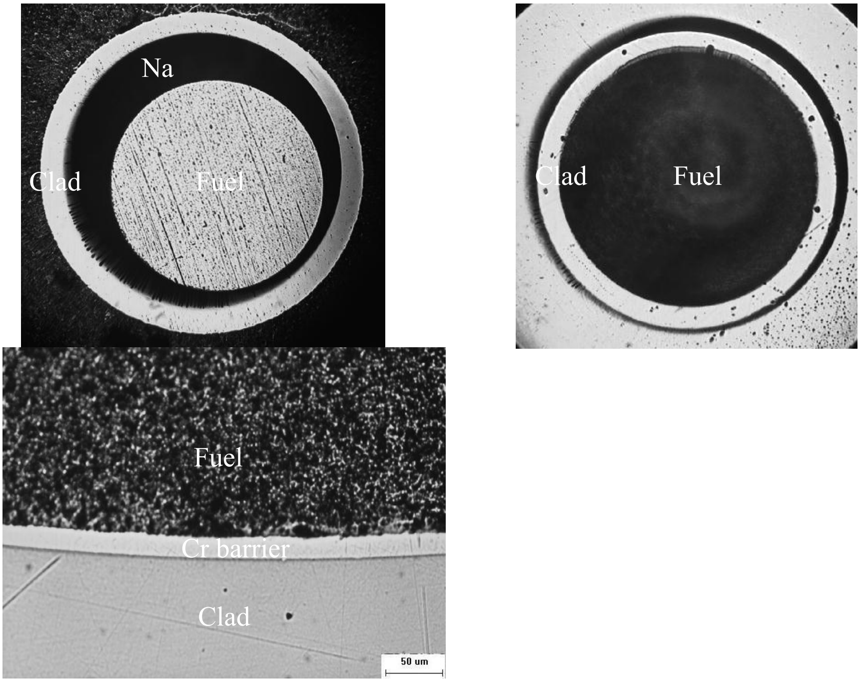


FIG. 6. Fuel irradiation test capsule in HANARO reactor.



Unirradiated fuel Irradiated fuel Irradiated Cr-plated fuel

FIG. 7. Cut surface of fuel rods.

3. SUMMARY

Metal fuel, U-Pu-MA-Ln-Zr, which uses TRU recovered through pyro-electrochemical processing of spent LWR fuel is being developed for a Generation-IV SFR. Since the fuel is highly radioactive, owing to the presence of MAs, and also contains the chemically active lanthanide elements, a simple fuel slug casting system which can control the volatile elements and can handle chemically active lantanide elements is being developed. FMS cladding is being developed for high bunup fuel and FMS cladding has been fabricated. Barrier technology to prevent interaction between metal fuel and cladding was investigated and Cr electroplating on the inner surface of the cladding was developed. Fabricated metal fuel rods, including Cr plated barrier cladding, were irradiated in the reactor and post-irradiation examination is currently ongoing.

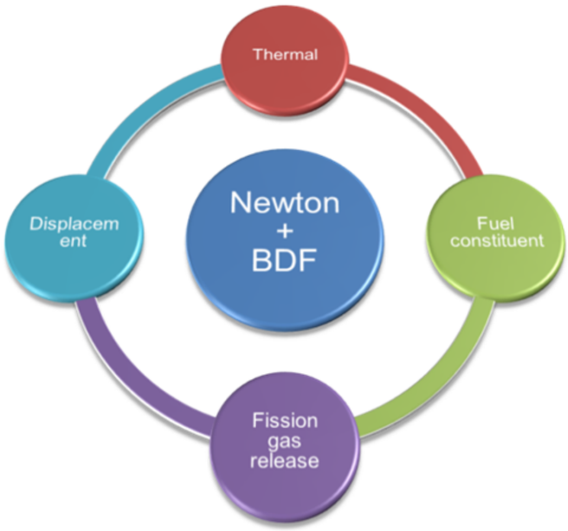


FIG. 8. Coupled approach in PUMA.

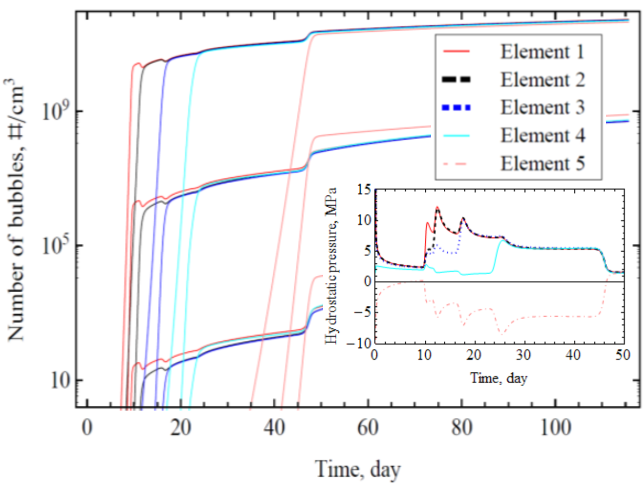


FIG. 9. Number of bubbles and hydrostatic pressure from PUMA.

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US DOE ADVANCED NUCLEAR FUEL DEVELOPMENT PROGRAMME OVERVIEW

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Abstract

The Advanced Fuels Campaign (AFC) has been given the responsibility to develop advanced nuclear fuel technologies for the Department of Energy (DOE) Fuel Cycle Research and Development Program using a science based approach, focusing on developing a microstructural understanding of nuclear fuels and materials. The science based approach combines theory, experiment and multiscale modelling and simulation to develop a fundamental understanding of the fuel fabrication processes and fuel and cladding performance under irradiation. The objective is to use a predictive approach to design fuels and cladding to achieve the desired performance (in contrast to more empirical observation based approaches traditionally used in fuel development). The AFC programme conducts research and development of innovative, enhanced, accident tolerant, next generation LWRs and transmutation fuel systems for sustainable fuel cycles. The major areas of research include enhancing the accident tolerance of fuels and materials, improving the fuel system's ability to achieve significantly higher fuel and plant performance, and developing innovations that provide for major increases in burnup and performance. The AFC programme is interested in advanced nuclear fuels and materials technologies that are robust, have high performance capability, and are more tolerant to accident conditions than traditional fuel systems. The scope of the AFC includes evaluation and development of multiple fuel forms to support the objectives described in the DOE Strategic Plan and the DOE's Office of Nuclear Energy Research and Development Roadmap. The word 'fuel' is used generically to include fuels, targets and their associated cladding materials.

1. INTRODUCTION

The mission of the Advanced Fuels Campaign (AFC) is to perform research, development and demonstration activities on advanced fuel forms (including cladding) to enhance the performance and safety of current and future reactors; enhance proliferation resistance of nuclear fuel; effectively utilize nuclear energy resources; and address the longer term waste management challenges. The mission also includes development of a state-of-the art research and development infrastructure to support the use of a 'goal oriented science based approach.' The scope of the AFC includes evaluation and development of multiple fuel forms to support the objectives described in the US Department of Energy (DOE) Strategic Plan and the DOE's Office of Nuclear Energy (DOE-NE) Research and Development Roadmap [1, 2]. The word 'fuel' is used generically to include fuels, targets and their associated cladding materials. Figure 1 provides a graphical depiction of the AFC structure. Research activities can be categorized as supporting the development of fuels for LWRs with accident tolerance, supporting the development of fast reactor metallic transmutation fuels, and development of technology, measurement techniques and methods that provide new capabilities for understanding the behaviour and performance of the nuclear fuel system.

2. NEXT GENERATION LWR FUEL DEVELOPMENT WITH ENHANCED ACCIDENT TOLERANCE

In 2011, enhancing the accident tolerance of LWRs became a topic of serious discussion in the United States of America. In the Consolidated Appropriations Act, 2012, Conference Report 112-75, the US Congress directed the DOE-NE to start developing nuclear fuels and claddings with enhanced accident tolerance.

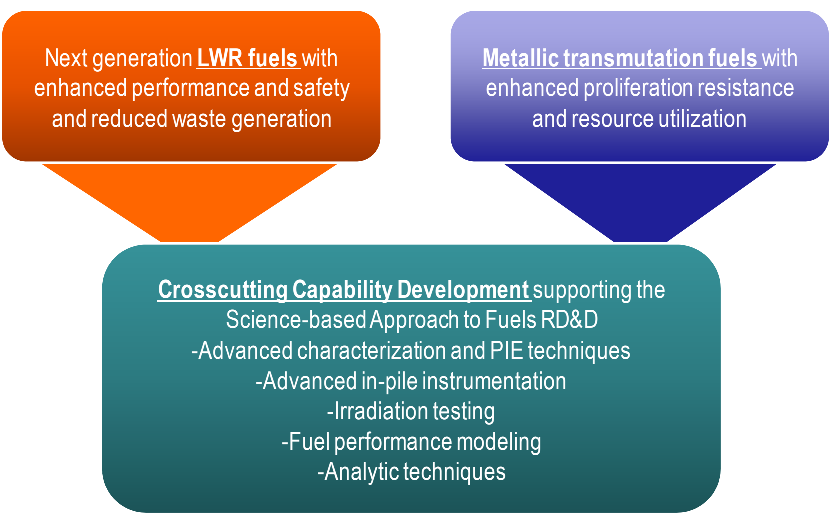

FIG. 1. AFC research development and deployment focus areas.

The Fuel Cycle Research and Development AFC has defined fuels with enhanced accident tolerance as those that, in comparison with the standard UO_2-Zircaloy system currently used by the nuclear industry, can tolerate loss of active cooling in the reactor core for a considerably longer time period (depending on the LWR system and accident scenario) while maintaining or improving fuel performance during normal operations, operational transients, as well as design basis and beyond design basis events.

Design objectives identified as potentially important in improving accident tolerance include: reduced hydrogen generation, improved fission product retention, improved cladding reaction to high temperature steam and improved fuel cladding interaction for improved performance under extreme conditions.

2.1. Hydrogen generation rate

Hydrogen buildup in the reactor vessel can lead to energetic explosions, such as those seen in the Fukushima events. Under a high temperature steam environment, it is not possible to totally avoid hydrogen generation. Rapid oxidation of cladding results in free hydrogen generation. This exothermic reaction increases the cladding temperature, which further accelerates free hydrogen generation. A related issue is the diffusion of free hydrogen into the unoxidized portion of the cladding, resulting in enhanced embrittlement and potential cladding failure.

A desired alternative would be a cladding material that resists oxidation or reduces the rate of oxidation, therefore resulting in a slower free hydrogen generation rate. Materials with lower heat of oxidation may be important in limiting the temperatures during an accident. Materials that are less susceptible to hydrogen diffusion may address the rapid embrittlement issue.

2.2. Fission product retention

Zircaloy cladding provides the initial barrier to the release of fission products in nuclear fuel. Upon cladding failure, retention of the fission products within the vessel is required to minimize releases to the environment. This includes both gaseous and solid fission products. Owing to the potential severity of fission product release to the environment, retention within the fuel is of the utmost importance. While total retention may not be possible, even partial retention (especially for highly mobile fission products) would be a substantial improvement.

The desired improvement would be to prevent melting or dispersion of the fuel by utilization of high temperature/strength materials. Additionally, fission product retention techniques or chemically linking the fission products in a fuel matrix may be options, as long as the concepts can tolerate high temperatures. Building additional barriers around the fuel to contain fission products (as a backup to containment provided by the cladding) may also be envisioned. An example of this concept is microencapsulated fuels.

2.3. Cladding reaction with steam

When exposed to steam at high temperature, there are multiple issues that need to be considered. As previously stated, the high temperature steam interaction with fuel cladding causes an exothermic oxidation reaction, resulting in hydrogen generation. In addition, this reaction deteriorates the structural integrity of the cladding, resulting in fission product release to the reactor vessel.

The design option would be to develop cladding materials with enhanced tolerance to radiation and oxidation under high temperature exposure while specifically considering mechanical strength and structural integrity at the end of life and when exposed to high temperature steam for an extended duration.

2.4. Fuel cladding interactions

In the event of cladding failure, fuel behaviour is important. The issues are fuel melting and relocation, as well as fuel dispersion into the coolant. Fuel cladding chemical interactions, fuel cladding mechanical interactions and fuel heating are important properties that must be understood during normal operation and accident conditions.

The design option would be to develop fuels with reduced fuel cladding chemical interactions and fuel cladding mechanical interactions, and with lower operating temperatures. Higher melting point and structural integrity at high temperatures (i.e. less dispersive) are also desired improvements.

2.5. Metrics for LWR fuels with enhanced accident tolerance

To demonstrate the enhanced accident tolerance of candidate fuel designs, metrics must be developed and evaluated using a combination of design features for a given LWR design, potential improvements, and the design of an advanced fuel/cladding system.

The aforementioned attributes provide qualitative guidance for parameters that will be considered for fuels with enhanced accident tolerance. It may be unnecessary to improve in all attributes and it is likely that some attributes or combination of attributes provide meaningful gains in accident tolerance, while others may provide only marginal benefits. Thus, an initial step in programme implementation will be the development of quantitative metrics.

The Fuel Cycle Research and Development AFC has embarked on an aggressive schedule for development of enhanced accident tolerant LWR fuels. The goal of developing such a fuel system that can be deployed in the USA's LWR fleet in the next 10–20 years supports the sustainability of clean nuclear power generation in the USA.

3. FAST REACTOR TRANSMUTATION FUEL DEVELOPMENT

Transmutation fast reactor fuel development in the AFC is focused on metallic fast reactor fuels and includes research of ceramic fast reactor fuels as a backup option. Transmutation fuels are those that contain the transuranic elements in addition to uranium. Near term research and development includes development of Zr based alloys with HT-9 cladding with longer term research targeting revolutionary concepts, including Mo based alloys, annular fuels, fuels containing fission product getters (lanthanides) and advanced steels with and without coatings or liners. The development process for these concepts requires the following:

Fabrication process development (fuels and cladding): Initially, laboratory scale fabrication will be performed but the process must be scaled to engineering scale demonstration. For transmutation fuels, the primary objective of fabrication process development is to minimize the losses during fabrication while enabling an industrial scale deployment.

Characterization of fresh fuels and cladding: The detailed characterization of the fresh fuel and cladding is important to (a) understand the initial conditions prior to irradiation, (b) to correlate performance parameters against the initial conditions, and (c) to assure the uniformity and reliability of the fabrication processes. The initial properties are essential in developing the fuel performance code.

Irradiation testing: Initially, rodlets in capsule will be tested in the ATR and HFIR (including rabbit testing). A limited number of fast spectrum tests is possible in the near term, primarily through international collaborations. Primarily relying on modelling and simulation, the thermal (or epithermal) testing done in the ATR will be correlated with the limited fast spectrum testing and legacy data. Transient testing of the fresh and irradiated fuel rodlets will also be required as part of the fuel qualification process. The observations made during transient testing are important for developing phenomenological models during postulated accidents.

Post-irradiation examination: The irradiated samples will be examined for quantifying the performance parameters. The performance parameters will be compared with the initial characterization results. The post-irradiation examination results, including legacy samples from previous irradiations, are essential to understanding the fuel behaviour under irradiation and to developing the fuel performance codes. This includes analysis of the data and comparative evaluation of various fuel types.

Out-of-pile testing: For metallic fuels prior to the qualification phase, out-of-pile testing will primarily consist of fuel–clad chemical interaction testing with fresh and irradiated fuel–clad diffusion couples.

3.1. Metallic fast reactor fuel development

Metallic based fast reactor fuels technology development is focused on research and development to gain a fundamental understanding of metallic fuels containing minor actinides. This includes developing low loss fabrication methods, determining burnup capabilities, and gaining a fundamental understanding of the phase, microstructure and chemical migration behaviour of metallic fuel constituents. Attention in these areas is focused on developing an understanding of the key phenomena affecting metallic fuel performance and behaviour in an irradiation environment. Some recent results obtained on the irradiation of advanced metallic fuel compositions is provided in Ref. [3] presented at this conference by Chichester et al. and efforts in advanced fabrication technique development provided in Refs [4, 5] by Fielding et al., also at this conference. The AFC programme is also pursuing development of advanced metallic fuel concepts for reliable performance to ultra-high burnup. A presentation of this activity is presented in Ref. [6] at this conference by Mariani, et al.

3.2. Ceramic fast reactor fuel development

Ceramic based fast reactor fuel technology development is focused on the fundamental understanding of oxide fuels. Priority is given to LWR fuels but some effort is given specifically to transmutation fuels. Key challenges include the development of reliable, low loss fuel fabrication methods and fuel technology development to enable major increases in fuel burnup and performance (reliability, power and safety) beyond current technologies. Activities fall within the following research and development categories: ceramic process modelling (sintering); U based fuel design, development and testing (international); mixed oxide fuel processing and properties (international); and technique development and reference materials. Some recent results obtained on the irradiation of advanced fast reactor ceramic fuel compositions is provided in Ref. [7] presented at this conference by McClellan et al.

3.3. Cladding materials development

The AFC includes significant efforts in developing cladding materials and technologies for high dose applications and advanced LWR cladding for enhanced accident tolerance in the following research and development areas: knowledge base development for high dose (up to 200 dpa) core materials irradiation data and advanced material development (advanced cladding materials and coatings/ liners to mitigate fuel cladding chemical interaction).

3.4. Fuel performance modelling and simulation

A significant part of a fuel development programme is the development of a fuel performance code accurately predicting the behaviour of the fuel system. The AFC programme supports the development of the BISON fuel performance code [8]. An example of the use of this code in simulating the constituent redistribution of metallic alloy fuels is provided in Ref. [9] presented at this conference by Unal et al.

4. SUMMARY

The Fuel Cycle Research and Development AFC is continuing to develop technologies for the LWR reactor fleet and is supporting nuclear fuel cycle closure using fast spectrum reactor technology.

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OUTCOMES ON OXIDE FUEL DEVELOPMENT FOR MINOR ACTINIDES RECYCLING

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Abstract

A state of the art review is given of minor actinide (MA)-bearing oxide fuel development for sodium fast reactors (SFRs) and accelerator driven systems (ADS). The homogeneous recycle option in SFrs, where small amounts of MAs are diluted in $(U,Pu)O_{2-x}$ driver fuels, emerges as a technically sound approach, reinforced by national and international programmes. Its technology readiness level is appropriate to implement irradiation tests from pin to bundle scale. Regarding the heterogeneous recycle option in SFRs, a comprehensive database regarding inert matrix fuels is available as the result of ~35 irradiation tests. The promising results gained with MgO, Mo and ZrO_2 matrices have to be completed by post-irradiation examinations on optimized fuel microstructures. On the other hand, a first step in the long term $(MA,U)O_{2-x}$ fuel development process is under investigation with MARIOS and DIAMINO tests in the HFR and OSIRIS, before the implementation of prototypical irradiation tests. For ADS, very informative feedback from inert matrix fuel developments has been completed by dedicated collaborative programmes, including major irradiations for the fuel performance assessment from HELIOS and FUTURIX-FTA experiments, whose post-irradiation examinations are under way.

1. INTRODUCTION

Minor actinide (MA) incorporation into the fuel is a prerequisite for Generation IV (Gen-IV) reactors and accelerator driven systems (ADS) to bring benefits in the disposal requirements by reducing the MA content in high level wastes. Since Am displays a strong gamma emission (and Cm a high neutron emission), the MA-bearing fuel fabrication process needs shielding, remote handling by robotic arms, and simplification as well as implementation of relatively dust free steps. Moreover, the high volatility of some Am compounds has to be managed during fuel fabrication as well as during irradiation where Am

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would be more readily redistributed within the fuel than other actinides. Finally, the harmful consequences of additional helium production during fuel irradiation (related to ^{242}Cm and ^{244}Cm formation in the ^{241}Am transmutation scheme) on fuel swelling, degradation of the thermal properties and high pressurization of the pins have to be prevented.

Based on historical experience and knowledge, oxide fuels have emerged in France as the shorter term solution to meet the Gen-IV assigned performance and reliability goals [1] and two main MA recycle options have been under consideration:

(i) The homogeneous mode, where small quantities (<3%) of MA oxide are diluted in the $(U,Pu)O_2$ of sodium cooled fast reactor (SFR) standard driver fuel, in order to limit the impact of MA addition on SFR core safety parameters (sodium worth void, Doppler effects, delayed neutron fraction) [2] and fuel cycle facilities.

(ii) The heterogeneous mode, where fuels are made with high MA oxide quantities (from 10 to 40%) mixed to:
- An inert matrix or UO_2, MA-bearing subassemblies being located in the periphery or in the blanket of as SFR core, respectively;
- An inert matrix as well as PuO_2, leading to high power density driver fuels for ADS.

The following sections give an overview of the development progress status for these four MA-bearing oxide fuel types.

2. HOMOGENEOUS RECYCLING IN SFRS: MA-BEARING DRIVER FUELS (MADF)

Besides the strong MA impact on SFR core neutronic parameters that limits MA content in MADF to ~3%, MA addition to $(U,Pu)O_{2-x}$ can significantly affect major fuel properties, i.e. melting temperature, thermal conductivity and oxygen potential, that are related to fuel behaviour and performance under irradiation as well as fuel fabrication. For example:

(a) A decrease in both melting point and thermal conductivity would obtain a higher central temperature and steeper fuel thermal gradient at the beginning of the irradiation, leading to a decreased margin of fuel melting as well as faster restructuring and thermal migration of U, Pu, Am, oxygen and volatile fission products (FPs), and therefore to an enlarged restructured area.

(b) An oxygen potential increase could lead to thermal changes and to increased migration of volatile FPs from the fuel restructured area to periphery, which would have consequences such as an earlier fuel–cladding chemical interaction and a faster fuel–sodium reaction in the case of cladding failure.
(c) Am volatilization during sintering is a concern.

Finally, the high helium production during irradiation could drive an increased FP release rate or enhance the fuel gaseous swelling favourable to fuel–cladding mechanical interactions.

National and international R&D programmes have been conducted over the past 25 years and many issues have been addressed by irradiations whose conditions are listed in Table 1.

SUPERFACT [3, 4] provided the first demonstration of the MA-bearing fuel good behaviour up to 6.5at.%, even if linear heat rate (38 kW/m) was slightly low compared to one for standard fuel. Fuels were fabricated through a dust free process implementing a sol-gel step [5]. Post-irradiation examinations (PIE) showed no significant evolution of the MA-bearing fuel microstructure compared to that for standard fuel. The restructuring features for MA-bearing fuels and standard fuels were almost the same. Neither Pu nor Am redistribution was found. Helium release was full whereas xenon and krypton behaviour were quite similar to the one in standard fuels irradiated under the same conditions. Finally, the cladding deformation was slightly higher (+0.4-0.5%) compared to standard fuels (+0.3%), whereas the cladding chemical corrosion depth was the same (<50 μm).

Additional irradiation data have been provided for the very beginning stage of irradiation (10 min and 24 h) by the Am1 first test performed on Am-bearing and Am+Np-bearing fuels. PIE [6] have shown that structural changes such as cracks, formation of lenticular pores and central void occurred within the first 10 min of the irradiation, when the linear heat rate was 43 kW/m. After 24 h of irradiation, the central hole diameter is significant and initial grains have clearly been replaced by columnar ones in the restructured area. Americium and Pu radial redistribution profiles, which are quite similar, show a moderate migration of both elements towards the central hole, whereas Np distribution remains flat in the overall radial cross-section. Under no circumstances was any sign of fuel melting found. Finally, regarding out-of-pile properties, experimental results on Am1 unirradiated fuels show that the melting temperature of MA-bearing fuels is slightly affected by MA addition (-3 K/MA%) and that the thermal conductivity decrease is moderate (≤7%) for temperature and O/(U+Pu+MA) ranges of 700–2900 K and 1.95–2.00, respectively [7, 8].

TABLE 1. LIST OF MADF IRRADIATION TESTS COMPLETED AND YET TO BE DONE

Test	SUPERFACT	Am1	AFC-2C and AFC-2D	SPHERE	GACID
Date	1980s	2008	2008–2010	2013	2017
Participants	CEA/JRC-ITU	JAEA	DOE-INL	FP7-FAIRFUELS	GACID-PMB
Reactor	PHENIX	JOYO	ATR	HFR	MONJU
Fuel form	Pellets	Pellets	Pellets	Pellets and spherepac	Pellets
Am content	2%	2–5%	2%	4%	3%
MA compounds synthesis process	Sol-gel	Powder metallurgy		Gelation and Am infiltration	Co-precipitation
Burnup	6.5at.%	10 min and 24h	8 and 19at.%	Ready for irradiation	Under preparation
Linear heat rate (kW/m)	~38	43	<30	~30	-

The campaign AFC-2C and AFC-2D [9] implemented to investigate MADF irradiation behaviour at high burnup in the ATR (using neutron shields to cut the thermal flux) is ongoing. The fuels were irradiated at 8 and 19at.% HM [10]. PIE, which is under way, will give first answers to issues related to fuel–cladding mechanical interaction and fuel–cladding chemical interaction.

Even if MA-bearing fuel pellets have been preferred so far, the implementation of the spherepac technology for the fabrication of MA-bearing fuel shaped as beads could be appropriate since it would lead to a significant simplification of the fabrication process with the elimination of steps that involve fuel powders (and dust). The irradiation SPHERE [11] that is due to start very soon in the HFR emerges as a first of a kind since irradiation behaviour of spherepacked and pelletized stacks of MADF will be compared.

Finally, the GACID project (2007–2025) [12], which is conducted within the Gen-IV, will provide the next key data required for the homogeneous recycling demonstration in SFR systems, with the implementation of irradiations in MONJU from one pin in 2017 to a bundle of pins later.

3. HETEROGENEOUS RECYCLING

An alternative way to the homogeneous recycling strategy consists of decoupled standard fuel/MA management by loading high MA quantities (up to 40%) in:

(a) An inert matrix or UO_2, for MA recycling in the SFR core periphery and radial blanket respectively;
(b) An inert matrix as well as PuO_2, for MA recycling in a double strata of ADS.

Thus, SFR core management and safety parameters remain almost unaffected. MA-bearing fuel fabrication and reprocessing will be implemented in highly shielded facilities of small capacity. Finally, in the case of $(MA,U)O_2$ fuels, the well known PUREX dissolution process will take part of the reprocessing route and UO_2 will be the main secondary stream.

Nevertheless, solutions have to be found to manage the following key issues:

(a) Fuel behaviour under irradiation (normal and off-normal) with concerns such as:
- Intense fast neutron flux leading to heavy irradiation damage that can drastically affect the fuel microstructure and properties;

- High helium production at moderate temperatures of irradiation (500–1500°C) that are favourable to excessive fuel swelling rates.

(b) High gamma and neutron doses as well as heat production complicating the fabrication process and the fuel treatment step.
(c) High decay heat level making in-core and out-of-core assembly handling difficult.

The following sections give a progress status on developments for each fuel type.

3.1. Inert matrix fuels

Basically, inert matrix fuels (IMFs) consist of single phases (i.e. a solid solution) or composites (MA oxide particles homogeneously distributed in the inert matrix), the MA content ranging from 10 to 40% according to the specifications.

An extensive R&D programme has been conducted within the framework of the French Acts dated 30 December 1991 and 28 June 2006, related to waste management optimization. As a result of national, European and international projects, a comprehensive database is now available on the ~35 irradiation tests (see Table 2) performed according to the methodological approach described hereafter.

After a preliminary screening of inert matrix candidates [13] based on past experience and knowledge of material and fuel science, irradiations have gradually been implemented to investigate:

(a) The effect of neutron flux and temperature on the selected support candidates, through EFFTRA-T2 & –T2bis and MATINA 1 [14].
(b) Both neutron flux and FP impacts using IMF surrogates (UO_2+IM, PuO_2+IM) via: MATINA-1A [14], EFFTRA-T3 [15], BORA-BORA [16], THERMET [17] and TANOX [18].
(c) The coupled effect of neutron flux, FPs and helium by testing AmO_{2-x}-IM fuels through: EFFTRA -T4 [19] and -T4bis [20], as well as ECRIX-B and –H [21].
(d) The impact of fast reactor representative conditions and optimized IMF microstructure (tailored porosity, particle size) on the most promising candidates, in HELIOS [22], MATINA-2 and -3, CAMIX and COCHIX [14].

TABLE 2. CHARACTERISTICS OF IMF IRRADIATION EXPERIMENTS

Name	Reactor	Fuel composition	Irradiation objectives	Status
EFFTRA-T2, - T2bis	HFR	$MgAl_2O_4$, Al_2O_3, $Y_3Al_5O_{12}$, CeO_2	Neutron effect	Completed (1996–2006)
EFFTRA-T3	HFR	UO_2 + ($MgAl_2O_4$, Y_2O_3, $Y_3Al_5O_{12}$, CeO_2, MgO or $BaZrO_3$)	Neutron and FP effect on micro-dispersed microstructures	Completed (1997–2006)
MATINA-1, -1A	PHENIX	-$MgAl_2O_4$, Al_2O_3, $Y_3Al_2O_5$, MgO, TiN, W, Nb, V, Cr -UO_2 + (MgO & $MgAl_2O_4$)	Neutron and FP effect on micro-dispersed microstructures	Completed (1994–2009)
MATINA-2 and -3	PHENIX	UO_2 + MgO, $ZrYO_2$, MgO+$Ce_2Zr_2O_7$	Neutron and FP effect on micro- and macro-dispersed microstructures	Irradiation completed (2009) PIE to come
THERMHET	SILOE	$MgAl_2O_4$ + UO_2	Neutron and FP effect on micro- and macro-dispersed microstructures	Completed (1997–1999)
TANOX	SILOE	(Mo, $MgAl_2O_4$) + UO_2	Neutron and FP effect on micro- and macro-dispersed microstructures	Completed in 1996
BORA BORA (partly)	BOR60	PuO_2+MgO	Neutron and FP effect on micro- and macro-dispersed microstructures	Completed (1997–2008)
EFFTRA-T4 & T4bis	HFR	$MgAl_2O_4$+AmO_{2-x}	Neutron, FP and alpha particle effect on micro-dispersed microstructures	Completed (1996–2003)

TABLE 2. CHARACTERISTICS OF IMF IRRADIATION EXPERIMENTS (cont.)

Name	Reactor	Fuel composition	Irradiation objectives	Status
ECRIX-B & -H	PHENIX	$MgO+AmO_{2-x}$	Neutron, FP and alpha particle effect on micro-dispersed microstructures	Irradiation completed (2008) PIE to come
CAMIX	PHENIX	$(Am,Zr,Y)O_2$	High burnup (25at.%)	Irradiation completed (2009) PIE to come
COCHIX	PHENIX	$MgO+(Am,Zr,Y)O_2$	High burnup (25at.%)	Irradiation completed (2009) PIE to come
HELIOS (partly)	HFR	$MgO+Am_2Zr_2O_7$ $(Am,Zr,Y)O_2$ $(Am,Zr,Y)O_2 + Mo$	FR conditions, optimized microstructure and temperature	Irradiation completed (2010) PIE in progress

Based on current knowledge and specifications, MgO, Y-ZrO_2 and Mo emerge as promising matrices as they are resistant to both neutron and FP damage [23]. Through the ECRIX-H PIE results, which have highlighted the satisfactory behaviour of a composite of 16.5% $AmO_{1.62}$ micro-dispersed in a MgO support for an Am fission rate of 25at.% [21, 24], MgO has become a primary candidate. This demonstration will be completed and extended to Y-ZrO_2 and Mo through the execution of PIE that will assess the impact of:

(a) Am oxide in Y-ZrO_2 and Mo matrices (HELIOS – pin 2 and pin 4);
(b) A tailored open porosity expected to promote helium release (HELIOS – pin 1);
(c) A macro-dispersion of fissile particles expected to limit FP damage in matrices (MATINA-2 and -3, COCHIX);
(d) High temperature operating conditions (>1100°C) expected to promote He release and inert matrix damage recovery (MATINA-2 and -3, CAMIX, HELIOS – pin 3).

3.2. MA-bearing blanket fuels

The MA-bearing blanket (MABB) fuel recycle concept consists of irradiating (U, MA)O_{2-x} fuels (10≤MA≤20%) in SFR radial blankets over periods two to three times longer than for standard fuels. MABB fuels operate at moderate temperatures (500–1500°C) that are likely to cause significant swelling as helium production can reach 4–7 cm^3/g_{fuel}, depending on the recycling scenario.

Even if UO_2 behaviour under irradiation is well known, experimental data on MABB remains scarce, with the exception of experience gained from SUPERFACT in the 1980s, where pellets of $U_{0.6}Am_{0.2}Np_{0.2}O_{1.926}$ prepared via a sol-gel route were irradiated up to 4.08at.% at high temperature (>1700°C), which lead to the complete release of helium during irradiation, a highly porous fuel microstructure and the occurrence of a mechanical interaction between the fuel and the cladding [3, 4].

A comprehensive R&D programme of MABB fuel qualification started in 2008 [25]. It includes, as a first stage, two separate effect irradiation tests, MARIOS and DIAMINO, that aim to investigate helium behaviour and fuel swelling as a function of temperature, MABB microstructure and He production rate (see Table 3). For both irradiations, fuels shapes, pins, sample holders and irradiation devices were specifically designed to get an accurate control of the temperature, to provide a flat intra-pellet temperature distribution and to allow free swelling of the fuel [26].

TABLE 3. EXPERIMENTAL REQUIREMENTS FOR MARIOS AND DIAMINO SEPARATE EFFECT IRRADIATIONS

Am content	Microstructure	Targeted temperature (°C)			
		600	800	1000	1200
15%	Standard	DIAMINO	DIAMINO	MARIOS	MARIOS
15%	Optimized	DIAMINO	DIAMINO	MARIOS	MARIOS
7.5%	Standard	—	DIAMINO	—	—
7.5%	Optimized	—	DIAMINO	—	—

The MARIOS irradiation, which was designed, prepared and performed within the framework of the FP-7 FAIRFUELS project [11], was achieved in May 2012 after ~304 d of irradiation in the HFR. Temperatures and He production rates were consistent with the requirements [27, 28]. PIEs are currently under way within the framework of the FP-7 PELGRIMM project [29]. The preparation of the DIAMINO experiment [26] is almost complete and the irradiation in OSIRIS should start by 2014. The fabrication of MARIOS and DIAMINO samples was performed in teleoperated shielded cells of the ATALANTE CEA facility by implementing new flowsheets to provide the tailored microstructures [30, 31] requested by the specifications.

The next step in the MABB fuel qualification rationale consists of the semi-integral experiment MARINE within the framework of the FP-7 project PELGRIMM [29]. The MARINE test will be the matching piece to the SPHERE irradiation on MADF currently under way within the FAIRFUELS project. Indeed, MARINE will investigate the behaviour of pelletized and spherepack $(U,Am)O_{2-x}$ fuels stacked in two instrumented (online pressure measurement) small pins in the HFR. The MARINE test is expected to be in-pile in 2013 for ~390 d.

Finally, even if reference routes for MABB fabrication are still based on powder metallurgy (MARIOS & DIAMINO) or the sol-gel process (MARINE), results gained on $(U,Am)O_2$ co-conversion by oxalate precipitation and on ion exchange resin developments offer promising prospects [32].

3.3. ADS fuels

Am-bearing driver fuels for ADS are highly innovative in comparison with those used in SFR cores: they are not fertile so as to improve the transmutation performance, and they contain high volumetric contents (~50%) of both MA and Pu oxides diluted in an inert matrix. Even if their behaviour under irradiation is quite unknown, irradiation tests on IMF (see Table 2) emphasized the major roles played by irradiation conditions (including temperature), helium production and material swelling due to microstructure modifications, amorphization, helium accumulation, etc. To go further, three irradiation tests were successfully implemented within the FP-6 European Project IP-EUROTRANS (2005–2010) to investigate:

(i) ADS type fuel behaviour under ADS representative conditions with FUTURIX-FTA in PHENIX [33];
(ii) Helium behaviour versus temperature and microstructure with HELIOS in the HFR [34];
(iii) Helium buildup and release mechanisms versus temperature in ^{10}B (as an Am surrogate) doped matrices with BODEX test in the HFR [35].

FUTURIX-FTA compositions (see Table 4) specifically address ADS type fuels whereas HELIOS compositions range from ADS type fuels to IMF for SFR. The Am content ranges from 0.2 to 1.9 g/cm^3 in FUTURIX-FTA pellets, whereas it is about 0.7 g/cm^3 in most HELIOS pellets. In BODEX, the three promising IMF matrices, MgO, Mo and ZrO_2, were doped with ^{10}B compounds such that the helium amount after two HFR irradiation cycles is similar to HELIOS one.

The highly radioactive materials were fabricated at laboratory scale in two steps. Americium particles were first synthesized using two processes: an oxalic co-precipitation route for MgO-CERamic/CERamic compounds [36] and a combination of external gelation and infiltration methods for Mo-CERamic/METallic composites and homogeneous compositions (i.e. ZrO_2 matrix) [37]. The following steps were based on conventional powder metallurgy and were similar for all compositions except HELIOS CERCER fuel, whose porosity was tailored to remain open in order to allow helium to escape [38].

TABLE 4. FUTURIX-FTA AND HELIOS FUEL COMPOSITIONS

FUTURIX-FTA	HELIOS
Pin 5: $Pu_{0.80}Am_{0.20}O_{2-x}$ + 86vol.% Mo	Pin 1: $Am_2Zr_2O_7$ + 80vol.% MgO
Pin 6 : $Pu_{0.23}Am_{0.24}Zr_{0.53}O_{2-x}$ + 60vol.% Mo	Pin 2: $Zr_{0.80}Y_{0.13}Am_{0.07}O_{2-x}$
Pin 7: $Pu_{0.5}Am_{0.5}O_{1.88}$ + 80vol.% MgO	Pin 3: $Pu_{0.04}Am_{0.07}Zr_{0.76}Y_{0.13}O_{2-x}$
Pin 8: $Pu_{0.2}Am_{0.8}O_{1.73}$ +75vol.% MgO	Pin 4: $Am_{0.22}Zr_{0.67}Y_{0.11}O_{2-x}$ + 71vol.% Mo
	Pin 5: $Pu_{0.80}Am_{0.20}O_{2-x}$ + 84vol.% Mo

The FUTURIX-FTA irradiation ended in February 2009 after 235 equivalent full power days (EFPD) in-pile. For CERMET fuels, maximum linear heat rates were 13 kW/m; maximum burnups were 18at.% (pin 5) and 13at.% (pin 6). For CERCER fuels, maximum linear heat rates were ~9 kW/m; maximum burnups were 9at.% (pin 7) and 6at.% (pin 8). PIE will be performed in the near future within the FP-7 project FAIRFUELS.

The HELIOS irradiation test ended after an irradiation time of 241 EFPD in-pile. The use of internal thermocouples located in pins 2 and 3 as well as thermocouples surrounding the cladding was successful. Consequently, approximate fuel central temperatures in pins 2 and 3, as well as external cladding temperatures of all pins, were accurately recorded during the irradiation [34]. PIEs are currently under way within the FP-7 project FAIRFUELS.

Regarding BODEX, the irradiation was performed at two temperatures: 1073K and 1473K. The experiment included neutron fluence detectors in all the capsules as well as on-line pressure and temperature measurements for capsules with MgO and Mo pellets heated at 1473K. PIEs have provided major results [35]: helium release is 3 to 4 times lower for Mo compared with ZrO_2 and MgO; swelling is (as always) very low (≤4%) for Y-ZrO_2 and remains manageable (≤9%) for Mo and MgO.

Besides the in-pile experiments, studies on ADS fuels within the FP-6 project EUROTRANS, motivated by assessing the industrial practicability for actinide transmutation have provided a wide range of results [39]. At the present time, owing to major issues expected in ZrO_2 reprocessability, ZrO_2 based fuels are considered as a backup solution. Both MgO-CERCER and Mo-CERMET fuels emerge as primary candidates. PIE results on HELIOS and FUTURIX-FTA pins nevertheless remain decisive for the next steps of the assessment.

4. CONCLUSION

Transmutation of MAs has been thoroughly investigated since the 1980s. It started by focusing on the homogeneous recycling in SFRs and the very first irradiation experiment, SUPERFACT, was performed in PHENIX. About 10 years later, the development of IMF and ADS fuels was initiated within the framework of the French Act dated 30 December 1991. An extensive irradiation programme was performed in the SILOE, PHENIX and HFR reactors to select the most promising inert matrices and fuel designs. Since 2006, besides developments on the homogeneous recycling, IMF and ADS fuels, an alternative scenario where $(MA,U)O_2$ fuels are located in SFR radial blankets, has been under investigation.

As a consequence of an ambitious national programme, strong international partnerships and fruitful collaboration within the framework of European projects and the Gen-IV International Forum, a significant set of results on MA-bearing oxide fuels is now available.

The homogeneous recycle option in the SFR, whereby small amounts of MA are diluted in $(U,Pu)O_{2-x}$ driver fuels, emerges as a technically sound approach. Its technology readiness level [40] is appropriate to implement irradiation tests from full scale pin to pin bundle within the framework of the GACID — Project Management Board.

Regarding the SFR heterogeneous recycle option in IMF, a comprehensive database has been built. The next step will consist of the execution of PIEs on optimized fuel microstructures.

Regarding $(MA,U)O_2$ fuel developments, a first step in the fuel qualification long term process is under investigation with MARIOS and DIAMINO tests in the HFR and OSIRIS, respectively, before the implementation of prototypical irradiation tests in the HFR (MARINE by 2013–2014) and possibly in the ATR as a next step.

For ADS, the very informative feedback from IMF developments has been completed by dedicated collaborative programmes, including major irradiations on the fuel performance assessment that are HELIOS and FUTURIX-FTA experiments, the PIE results of which will be available in the near future.

The knowledge gained as a result of the lengthy efforts made on investigating fuel performance potential and limitation assessment will be used to achieve the industrial demonstration of Am transmutation in the future ASTRID reactor, considering the homogeneous recycling scenario (MADF fuels) as well as the heterogeneous mode (MABB fuels) [41].

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

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INDUSTRIAL MATURITY OF FAST REACTOR FUEL CYCLE PROCESSES AND TECHNOLOGIES

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Abstract

For more than sixty years, fast reactors have been using several fuel types, mainly oxide or metal, depending on the reactor and core design, as well as the fuel manufacturing and reprocessing capability. Future industrial deployment will require industrial capacity to extract plutonium from spent fuel, initially from other reactors, then from fast reactors themselves, as well as to manufacture Pu based fuel. This paper gives an overview of the industrial maturity of the different options, both for reprocessing and fuel manufacturing.

1. INTRODUCTION

With around 400 reactor-years of operating experience, about 20 fast reactors have already been operating, with thermal power ranging from a few MW up to several thousands MW. Table 1 presents a brief summary of the fast reactor, including an estimate of the total thermal energy produced.

Additionally, two reactors are under construction, the BN-800 in the Russian Federation and the PFBR in India, and new projects are envisaged in China, France and the Republic of Korea in connection with domestic recycling strategy. Mixed oxide (MOX) fuel has been retained as a reference fuel for these projects.

2. FUEL TYPES DESCRIPTION AND MAIN CHARACTERISTICS

2.1. Oxide fuel

This fuel is generally made of plutonium and uranium oxide pellets, with a Pu/U+Pu ratio between 15 and 30wt%. Compared with metal fuel, its lower density, thermal conductivity and poor compatibility compared with sodium

TABLE 1. FAST REACTORS

Reactor name	Country	Main fuel type	MW (thermal)	Operating years (estimated)	Total thermal output (MW·y)
EBR1	USA	Metal	1.4	12	17
EBR2	USA	Metal U-Zr	62.5	31	1938
Fermi	USA	Metal U-Zr	200	9	1800
SEFOR	USA	MOX	20	3	60
FFTF	USA	MOX	400	13	5200
Rapsodie	France	UO_x/MOX	40	16	640
Phénix	France	MOX	563	36	20 268
SuperPhénix	France	MOX	3000	1	3000
KNK 2	Germany	Oxide	58	14	812
BOR-60	Russian Federation	UO_x/metal	55	43	2365
BN-600	Russian Federation	UO_x	1470	32	47 040
Monju	Japan	MOX	714	5	3575

TABLE 1. FAST REACTORS (cont.)

Reactor name	Country	Main fuel type	MW (thermal)	Operating years (estimated)	Total thermal output (MW·y)
Joyo	Japan	MOX	140	34	4760
DFR	UK	Metal	60	18	1080
PFR	UK	MOX	650	20	13 000
CEFR	China	UO_x/MOX	65	3	195
FBTR	India	Carbide/MOX	40	27	1080
BN-350	Kazakhstan	UO_x	750	27	20 250

coolant are less favourable features, whereas its higher melting temperature is an advantage.

Another point to consider is the larger amount of experience gained with reactors, both experimental and in nuclear power plants, as well as in reprocessing and manufacturing. As estimated from total thermal output from fast reactors, experience with oxide represents 93.7% of total fast reactor life, metal beeing at 5.5% and other fuel at less than 0.8%.

As an example, the former MOX fuel fabrication ATPu workshop (located at the Cadarache site in France) recycled 25 t of plutonium, which was used to produce 450 000 fuel pins, corresponding to more than 110 t of oxide. It is used mainly in the Phenix and Super Phenix reactor power plants.

2.2. Metal fuel

Among metallic fuels, U,Pu,Zr is the more frequent concept studied and tested in experimental fast reactors. The main advantages in comparison with oxide fuels are the following :

- Higher density, which is favourable to breeding ratio;
- Higher thermal conductivity (10 more in comparison with oxide) which gives better fuel behaviour against an unprotected transient accident (e.g. ULOF).
- Good compatibility with sodium coolant.

Despite these advantages, this type of fuel has some drawbacks:

- Swelling of the metallic matrix requires guaranteeing that the space between the metallic rod and clad is filled with Na (ensuring good heat transfer between the metallic rod and the clad).
- A risk of chemical interaction between the metallic matrix and the clad material which reduces the maximum linear power and therefore the size of the core (in addition to the need for increasing the gas plenum to cope with higher fission gas release).
- The fabrication process of such rods presents some additional difficulties due to pyrophoricity of the metal and the loss of Am during the fusion of material.
- Limited industrial experience on the treatment of this fuel.

2.3. Other fuel

Nitride and carbide fuels have been considered as they have some interesting characteristics for use in reactors. To overcome the current shortcomings of existing fuels, R&D organizations (CEA in particular) investigates potentially more effective fuels such as carbide and nitride ceramics.

Among these alternative fuels, most worldwide studies have been focused on carbide, carbonitride and nitride ceramics. The main interests for using dense uranium and plutonium fuel ceramics are the following:

- Good thermal conductivity leads to a low operating temperature at nominal power. Carbide and nitride can be considered as 'cold' fuels with an important margin to fuel melting. This behaviour improves core reactivity feedback coefficients. The Na coolant Doppler ratio is also more favourable and provides a more satisfactory dynamic behaviour and good passive safety.
- High heavy atom density favours a better breeding gain and therefore a smaller loss of core reactivity. Thus, longer irradiation cycles and/or a reduction in the number of control and shutdown rods are possible.
- Very good chemical compatibility of these dense ceramics with liquid sodium enables, in the case of clad failure, continued reactor operation with a broken fuel element until the next scheduled stop.

However, the design of carbide or nitride fuel element must take into account high swelling rates to ensure that the stress level in the cladding due to fuel–cladding mechanical interaction remains acceptable at high burnup. Swelling behaviour is certainly the most important drawback of these alternative fuels.

Although experiments have been made, no industrial or even pre-industrial facility has been constructed so far.

3. FUEL CYCLE DESCRIPTION

The fuel cycle for fast reactor fuels is composed of two major components: reprocessing and fabrication. Reprocessing initially applied to used fuel from other reactors, generally LWRs, then autorecycling of fast reactor fuel, to provide Pu. Fabrication to transform U (reprocessed U or depleted U) and Pu into fuel. Figure 1 shows fuel cycle with the two feed options.

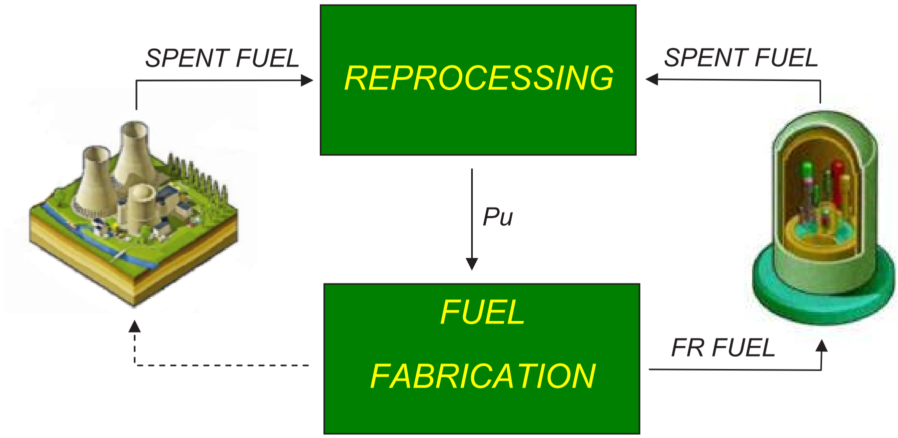

FIG. 1. Fuel cycle showing the two feed options.

Reprocessing plants typically use automated processes implemented in shielded cells to protect staff from radioactive emission of the spent fuel. Process maintenance inside the shielded cells is also performed remotely, using telemanipulators and cranes.

Owing to the use of plutonium, fuel fabrication plants for fast reactor fuel implement automated processes in gloveboxes, as the main hazard for staff is linked to contamination. Only the final stages of fuel assembly, after cladding tubes have been welded and tested, take place without gloveboxes.

In both cases, safety requirements and quality control MOX specifications put strong constraints on the design of such plants. In particular, criticality risk management always tends to limit the size of the batch (homogenized powder) and therefore thoughput. As a result, scaling up from laboratory to pilot and then to industrial scale has always been a very long and risky process, with some plants having never reached their planned capacity.

4. PROCESS DESCRIPTION

4.1. Reprocessing processes

4.1.1. Hydrometallurgy

Historically, the first process to be used for extracting plutonium from spent fuel is hydrometallurgy. This process is the only one to be currently used on an industrial scale in different countries, including France, India, Japan, the Russian

Federation and the United Kingdom. Worldwide actual capacity exceeds 2000 t/y. This process is used to extract uranium and plutonium mainly from uranium oxide fuel used in LWRs, but has also been used for MOX fuel from both LWRs and fast reactors. Although this has not been demonstrated yet, it is also meant to be usable for carbide type fuel, but not for metal or nitride fuels. It is currently based on the PUREX process, using tributhylphosphate as extractant. Several derivatives have been and are still being developed for efficiency improvement or minor actinides extraction.

More than 30 000 t of used fuel has been reprocessed using this process, including several tens of tonnes of MOX, from both LWRs and fast reactors (i.e. 540 Phenix irradiated fuel assemblies have been reprocessed providing 4.4 t of Pu recycled in new MOX fuels).

4.1.2. Pyroprocessing

Different types of high temperature, electrometallurgical process have been developed in different countries, which are generally referred to as pyroprocessing. Several laboratories or small pilot scale units have been developped and used so far, but none have reached industrial scale yet. Pyroprocessing is particularly suitable for metal fuel reprocessing and manufacturing, but has also been demonstrated at limited scale for oxide fuel. Although it is claimed to offer significant cost reduction over hydrometallurgy, the industrial feasibility of such processes for used fuel reprocessing has not been demonstrated.

4.2. Manufacturing processes

4.2.1. Oxide fuel process

Most of the oxide fuels used for fast reactors use the same design, based on uranium or mixed uranium and plutonium oxide (MOX) pellets inserted in stainless steel tubes. Although Pu content and detailed specification differs between LWR and fast reactor fuels, the manufacturing processes for obtaining pellets from uranium and plutonium powders are very similar. After one or two stages of mixing and/or milling, the oxide powder is pressed into pellets that are then sintered. The MELOX plant (MOX for LWRs) is currently operating in France, while others are under construction or planned in Japan, the Russian Federation and the United States of America. Total production to date amounts to a few thousand tonnes, of which several tens of tonnes is attributable to fast reactors.

Another process that has been developed, particularly in the Russian Federation, for manufacturing oxide fuel without pellets is called Vibropac. In this case, oxide powders with calibrated granulometry are directly introduced into the cladding tubes, using vibration to reach the desired density.

4.2.2. Metal fuel process

In this case, fuel is made from long bars of metallic alloy, generally U,Pu,Zr. The metal, which melts at around 1300°C, is poured into quartz tubes used as as moulds (injection casting) under an argon atmosphere. The tubes are then broken to recover the fuel bars, which are cut to the required length and introduced into the cladding tubes. Sodium metal is used to fill the gap between the pellets and tubes, and improves thermal conductivity.

The microstructure is very homogeneous and no grinding is required to fulfil the dimensional requirements. Figure 2 illustrates the main steps of the process.

5. CONCLUSION

Several hundreds of tonnes of fast reactor fuel have been fabricated to date, using different processes, with metal and oxide being by far the most developed forms. Plutonium recovery, from both thermal reactors and fast reactors, has also been conducted on an industrial scale in different countries. For these reasons, the fast reactor fuel cycle can be considered as being relatively mature from an industrial standpoint, and highly developed, depending on the fuel type.

Based on the industrial experience already gained on fast reactors and the significant synergies existing with the LWR fuel cycle, the oxide route is by far the more mature. As any other fuel type would have to face a very significant development cost and time, it should provide a very clear advantage, from a safety and/or cost point of view in particular, to justify the corresponding investment.

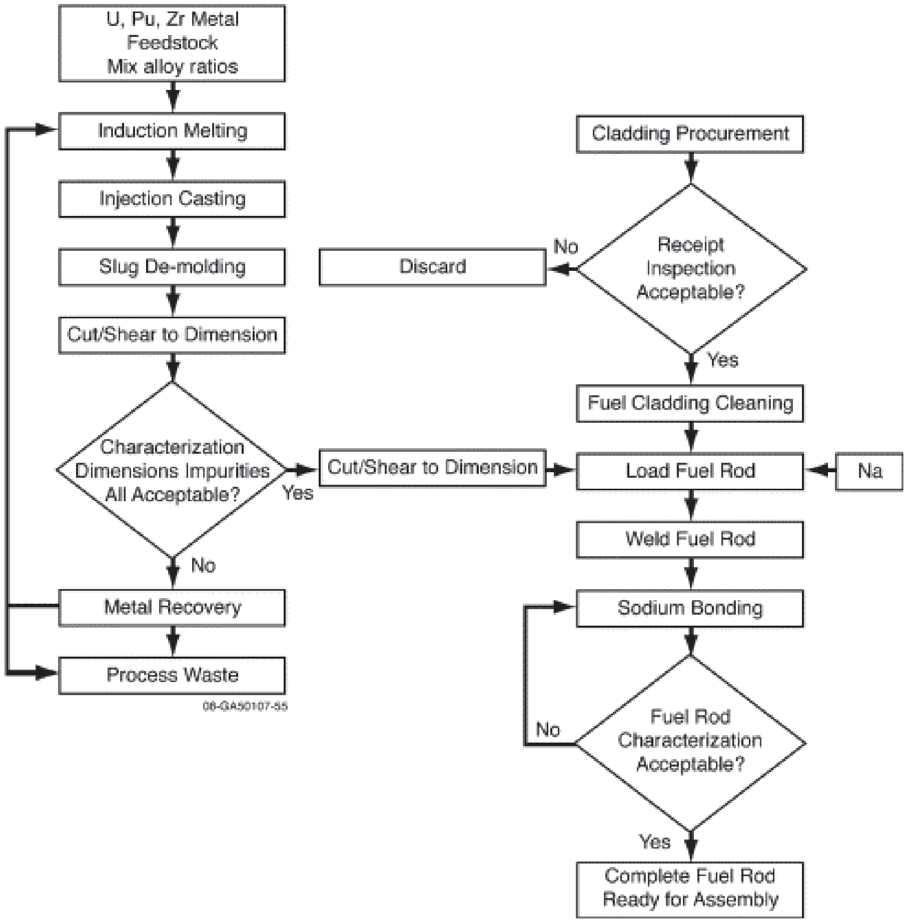

FIG. 2. The metal fuel production process.

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CHEMICAL AND TECHNOLOGICAL ISSUES OF THE FAST REACTOR FUEL CYCLE

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Abstract

The most important problems of closed fuel cycle with the fast reactors are safety, environmental attraction (acceptability) and economic efficiency. The existing spent nuclear fuel (SNF) reprocessing technologies and mixed uranium-plutonium fuel fabrication technologies do not satisfy all these requirements. The drastic improvement of existing technologies and/or development of new ones are necessary for closing the fast reactor nuclear cycle. The non-aqueous technologies can be used for the reprocessing of fast reactor SNF with low cooling time and high burnup. However, success cannot be achieved without the reduction of the costs of all stages of SNF reprocessing and fuel fabrication. The common solution to these problems is not clear; there are many different approaches that can be used for solving separate small tasks. The review of the main directions of studies in the fast reactor fuel cycle is given in this paper.

1. INTRODUCTION

The most important problems of the closing of the fast reactor nuclear fuel cycle are ensuring safety, environmental attractiveness and attaining economic efficiency. The existing industrial technologies for reprocessing spent nuclear fuel (SNF) and the fabrication of mixed uranium-plutonium fuel do not make it possible to solve the aforementioned problems fully. Only radical improvement of the existing technologies or the development of new ones will make it possible to close the fuel cycles of fast reactors.

Analysis of the existing technologies and those under development in the area of the fast reactor nuclear fuel cycle indicate that unsolved chemical and technological problems can be divided into three primary categories:

(i) Reducing the duration of the ex-core fuel cycle for decreasing amounts of SNF and nuclear materials in interim storage and increasing fuel burnup;
(ii) Reducing expense due to SNF reprocessing and fabrication of nuclear fuel by shortening (reducing the number of operations) the technological process, reducing the volumes of secondary and non-technological waste and increasing the service life on the equipment.

(iii) Reducing expenditure on handling radioactive waste via transmutation of long lived actinides, recycling of construction materials, and use of degradable reagents during processing.

Furthermore, the transition to a closed fast reactor nuclear fuel cycle introduces supplementary questions connected with the high content of fissile material in SNF and refabricated fuel. This paper will consider how the aforementioned problems affect the technologies for the reprocessing of fast reactor SNF.

2. HIGH BURNUP AND LOW COOLING TIME

As of today, the average burnup of the reprocessed uranium or mixed (uranium and plutonium) oxide SNF of thermal neutron reactors amounts to approximately 55 and 45 GW·d/t, respectively, while a burnup of 96 GW·d/t and more is forecasted for mixed uranium-plutonium SNF from fast reactors [1]. Increasing the burnup by nearly twice leads to an almost proportional increase in the concentration of fission products, many of which present a serious problem from the point of view of precipitate formation during SNF reprocessing with traditional hydrometallurgical methods based on the PUREX process [2]. Increasing fission product content inevitably leads to the formation of precipitates either of molybdenum and zirconium if the solutions are of low acidity or strontium and barium if the solutions are of high acidity [3]. High plutonium content in the solutions, an average of 16% from the sum of uranium and plutonium [1], just as inevitably leads to capture of plutonium by the precipitates and the need to conduct special operations to clear the precipitates, which has a negative effect on economic efficiency even without the hardly inexpensive process of reprocessing SNF. Potential reduction via dilution of the solutions leads to an increase in the amount of solution to be processed and, accordingly, the volumes of high and intermediate level wastes. It is unlikely that such a solution will turn out to be economically efficient, even if the technology of voloxidation is used for preparatory distillation of the tritium [4], or crystallization to forego an organic extraction [5], even counting the significant reduction in the number of operations associated with hydrometallurgical processing for oxide SNF from thermal [4] and fast reactors [6]. That is, as the burnup of SNF grows, the attractiveness of traditional hydrometallurgical technologies is reduced, including low water and low waste supercritical fluid extraction [7, 8].

The necessity of deducing the duration of the ex-core fuel cycle is not obvious. However, short duration of the ex-core fuel cycle allows decreasing the amount of SNF in the storage pool and (in principle) to reducing the use of a storage pool. It should be remembered that not only the reactor core but also the storage pool require the safety and security systems. More than that, in the case of short duration of the ex-core fuel cycle, the amount of nuclear materials in all fuel cycles will also be reduced. This is an advantage of the short duration of the ex-core fuel cycle. The drawback is a management with low cooling time SNF, which has a high heat emission. The high heat emission could make a hydrometallurgical technology unsuitable for reprocessing of SNF with low cooling time and require use of a pyrochemical (dry) technology [9].

Dry SNF reprocessing technology has been developed for many years, including technology in the context of reprocessing dense fast reactor fuel [10]. It should be noted that some of the pyroelectrochemical processes under development [10, 11] envisage, as a first step, the redevelopment of oxide to metal (see Fig. 1), which makes it possible to use electrorefining technology in the future [12]. Dry reprocessing technologies also include gas fluoride technology [13], the advantages of which are obvious in the case of thermal neutron SNF reprocessing (the final product of reprocessing (UF_6) is ready for enrichment) and it is precisely the production of uranium in the form of hexafluoride which is suitable for enrichment. These advantages are not obvious for fast reactor SNF reprocessing.

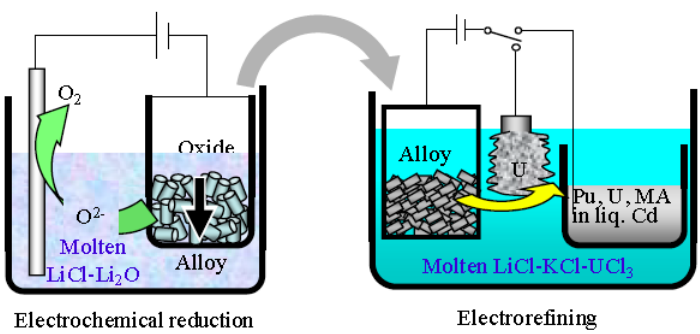

FIG. 1. Diagram of pyrochemical process for reprocessing mixed oxide (MOX) fuel.

Another factor that can lead to reduction of expenditures during reprocessing of fast reactor SNF with low cooling time and with high burnup is the use of various combined technologies, based on both purely 'dry' methods [14 ,15] and on a combination of 'dry' and 'aqueous' processes [16]. Each of these methods uses non-aqueous primary operations (Fig. 2), which make it possible to separate actinides from fission products without using aqueous solutions, which make it possible to work with fuel with a low cooling time. In this way, it is obvious that low cooling time and high burnup makes it necessary for the non-aqueous operations be used to lead the processing process.

3. HIGH FISSILE MATERIAL CONTENT

Another distinguishing feature of reprocessing fast reactor SNF is the high fissile material content in the SNF. As of today, only France [17] and the Russian Federation [18] have experience in the industrial reprocessing of this kind of fuel. It is obvious that the problems connected with ensuring nuclear safety once again make it necessary to use 'dry' SNF reprocessing methods. It must be noted, however, that hydrometallurgical methods are also suitable for work with large quantities of fissile material, but require the use of either dilute solutions or ring

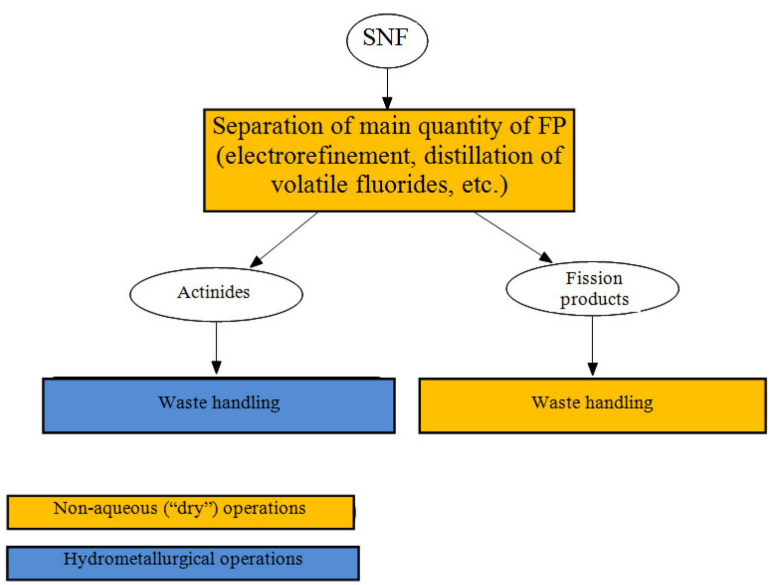

FIG. 2. Flowchart of primary operations of combined technologies for processing of SNF with high burnup and low cooling time.

devices, which have a negative impact on the economy of the SNF reprocessing owing to growth in the volume of devices and solutions.

4. REDUCING THE NUMBER OF OPERATIONS AND THE VOLUME OF WASTE

Attempts to achieve reduction in expenditures by reducing the number of operations have primarily been made by developers of hydrometallurgical technologies [4, 6]. This is probably connected with the accumulated experience gained in the industrial use of hydrometallurgical radiochemical production sites and an understanding not only of the problems themselves, but also of the effect of the proposed solution on the process as a whole [19]. Furthermore, specifically for hydrometallurgical technologies intended to prevent the formation of large quantities of aqueous and organic radioactive waste, there have been proposals for a transition to precipitant processes on a modern level [20, 21], of course, and a transition to the use of inorganic carriers for sorption processes [22] and the use of direct denitration to produce uranium and plutonium powders [23, 24], and much more. Specifically for hydrometallurgical and combined technologies, research is also under way on the corrosiveness of media to facilitate selection of more durable construction materials [25, 26] and the possibility of using diluted solutions [27] as part of the well-known CHON concept. Transferring the ideology of reducing expenditures not only by switching technology, but also by combining heterogeneous processes into one, increasing the service life of equipment, using more durable carriers for sorption and adsorption. The principle of “economy everywhere and in all things” to the development of ‘dry’ technologies is one of the tasks of researchers, technologists and designers, of course, after having first solved all the safety and security issues, which are not discussed in this paper.

5. MINOR ACTINIDE MANAGEMENT

No one doubts that handling minor actinides is a key issue, and not only management with fast reactor SNF but the entire closed fuel cycle [28–30]. It is no less obvious that minor actinide transmutation in fast reactors is the most promising method for handling minor actinides (Fig. 3). However, the question of whether the transmutation should be homogenous or heterogeneous has not be solved unambiguously, just as the question of the final fate of the curium has yet to be resolved.

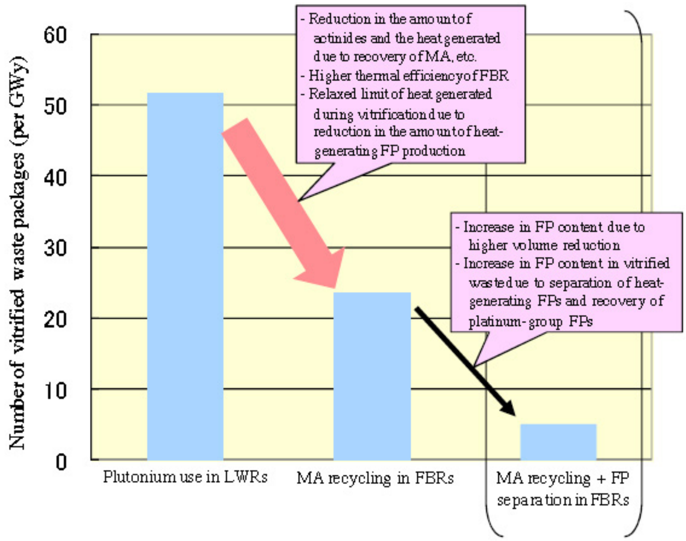

FIG. 3. Reduction effect of minor actinides recycling [34].

The resolution for these issues affects not only the structure of the fuel cycle, but also the choice and development of technology for the SNF reprocessing and the requirements for the radwaste solidification and final disposal. Wide-scale research is under way, as a minimum in Europe (the ACSEPT programme) [31], in the area of partitioning by hydrometallurgical and pyroelectrochemical technologies. Unfortunately, not all of these technologies have been checked with real solutions and the tests were not always successful. The more promising ones are the EXAm [32] and SETFICS [33] processes. However, even for hydrometallurgical processes, these processes cannot be considered ready for industrial use. As such, the development of partitioning processes and their introduction into the industry is currently one of the most important tasks of radiochemical technology.

6. COMBINED (PYRO+HYDRO) TECHNOLOGY FOR FAST REACTOR SNF REPROCESSING IS A MAKESHIFT FOR RECYCLING NUCLEAR MATERIALS FOR FUEL PELLET PRODUCTION

As discussed earlier, all of the dry technologies that are currently being developed (pyroelectrochemical, gas fluoride, plasma, etc.) for SNF reprocessing make it possible to reprocess fast reactor spent fuel with cooling times of even less than a year, but at present, none of them can guarantee the production of a final uranium-plutonium product that is suitable for the fabrication of nitride or carbide pellets and cannot guarantee the return of 99.9% of the actinides to the fuel cycle. This is possible through the use of combined technology (pyro+hydro), which is based on a combination of pyroelectrochemical recovery of uranium-plutonium-neptunium and hydrometallurgical refining of that fraction.

The combined processing technology can be applied to oxide, nitride, carbonitride and metallic SNF of fast reactors. The combined (pyro+hydro) technology envisages the use of primary pyroelectrochemical operations that make it possible to reprocess fast reactor SNF with low cooling time (up to 0.5 years) and separate the greater part (up to 99%) of highly radioactive fission products generated from uranium, plutonium and neptunium separated from SNF for the refabrication of the fuel. Hydrometallurgical operations of the combined technology are intended for refining recycled components for the separation of americium and curium as well (see Fig. 4).

The hydrometallurgical process of combined fast reactor SNF reprocessing technology includes:

— Cathode deposit dissolution;
— Plutonium dioxide deposit dissolution;
— Off-gas cleaning;
— Clarification of solutions;
— Extraction and crystallization refining of the uranium-plutonium-neptunium mixture;
— Recovery of transplutonium elements from the raffinate of the extraction and crystallization process;
— Separation of Am and Cm;
— Evaporation processes;
— Production of U, Pu and Np oxides;
— Production of Am oxides (an option is the possible return of the americium nitrate solution to the production of U, Pu and Np oxides);
— Production of Cm oxides for long term storage;
— Management with intermediate level waste;
— Solidification of high level waste.

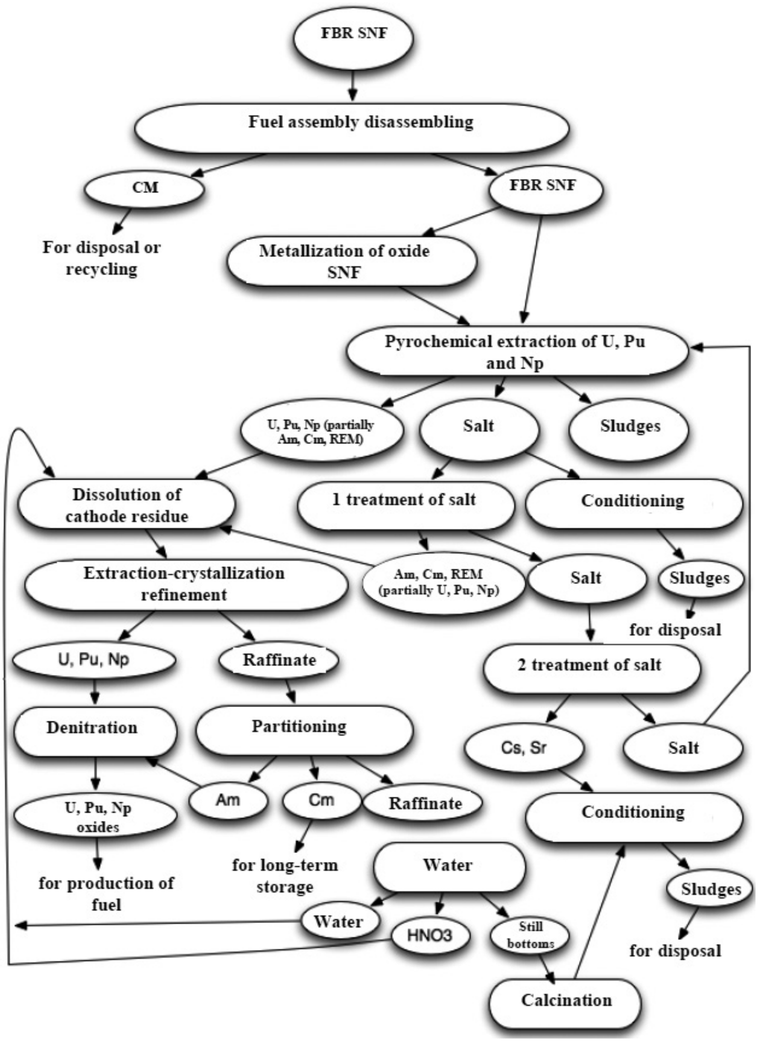

FIG. 4. Flowsheet of the combined (pyro+hydro) technology for fast reactor SNF reprocessing.

The proposed combined flowsheet is an example of a positive synergistic effect, when the combination of two methods leads to a combination of their positive characteristics and makes it possible:

— To reprocess fast SNF with high burnup and low cooling time, which will make it possible to reduce the volume of stored SNF and the quantity of plutonium in closed nuclear fuel cycle with fast reactors (reducing volumes of stored Pu);

— To process any type of fast reactpr SNF;
— To produce a uranium-plutonium-neptunium product of any level of purity, which makes it possible to use pellet technology to refine the fuel.

7. CONCLUSION

This work did not include consideration of the questions of management of high level waste, since the production of glass in ceramic melters and cold crucibles is an industrially viable process [35, 36], and the need to produce ceramic matrixes for waste is not obvious with regard to answering the question of the fate of minor actinides.

Thus, one of the possible ways to reduce the duration of the external fuel cycle is the use of aqueous technologies that make it possible to handle high burnup and low cooling time fast reactor SNF. However, without solving the cluster of programmes connected with lowering the expenditures due to handling SNF and radwaste, it is unlikely that the fuel cycle will reach economic efficiency. Currently, a global solution to the aforementioned problems is not obvious, and there are various trends associated with solving separate tasks.

For the industrial viability of a closed cycle with fast neutron reactors, the following technologies must be made industrially viable:

— Decladding;
— Handling volatile fission products (^{14}C, Kr, Xe, etc.);
— Recovery of transplutonium elements and separation of Am and Cm;
— Waste management and solidification of high level waste into forms that are suitable for final isolation;
— Handling construction materials;
— Technology for regenerating the medium (electrolyte, water, argon).

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SAFETY OF ADVANCED NUCLEAR FUEL CYCLES

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Abstract

Advanced nuclear fuel cycles require focused research and development programmes to support the implementation of a sustainable, more efficient use of resources and the safe and secure use of nuclear energy. Investigations in the fields of advanced fuel concepts and their safety relevant properties and irradiation behaviour, as well as separation strategies for closing the fuel cycle, are presented. The results are integrated within the overall perspective of the 'Sustainable Nuclear Energy Platform', and the related European Commission research programmes, and, at international level, within the Generation IV International Forum activities.

1. INTRODUCTION

The European Union 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 emissions reduction to meet the goals of the United Nations Framework Convention on Climate Change. After the Fukushima accident in 2011, a major emphasis in terms of public perception of nuclear energy has been given to safety aspects. However, owing to the long time periods involved with the storage of irradiated materials, and to the potential associated uncertainties, implementing solutions for the treatment and final disposal of nuclear wastes remains an equally important factor for obtaining public acceptance on the use of this source of energy.

In Europe, two main spent fuel strategies are being implemented, namely 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 disposal 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.

In order to achieve the goals of long term sustainability in nuclear energy, by using uranium resources in a much more efficient way, the nuclear research community has been strongly involved in exploring technical solutions to reduce the radiotoxic inventory of the nuclear waste (spent fuel, high level waste). The prevailing concept is the further separation of the minor actinides (MAs) from the spent fuel during the reprocessing, generally called partitioning, and their destruction by re-irradiation with neutrons, transmutation. The technical feasibility of the partitioning and transmutation concept has been extensively studied for decades. From the reactor physics point of view, transmutation can be realized best in fast neutron systems, in which the minor actinide nuclides can be fissioned efficiently owing to the favourable fission-to-capture ratio for neutrons and the favourable neutron economy. Partitioning and transmutation in fast reactors could lead to a reduction by a factor of 10 in the size of radwaste disposal facilities [1], mainly due to a reduced heat load. A concomitant reduction in the radiotoxicity of the fuel from over 100 000 years to less than 1000 years could also be accomplished [2]. Besides the further development of the fast neutron system, the key technological issues currently addressed are the establishment of an efficient partitioning scheme for the minor actinides from the spent fuel, and the design of minor actinide fuel that can be introduced in a reactor without adverse effects on the system safety. It is thus clear that the separation technology and fuel technology for minor actinides are critical for the further development of partitioning and transmutation strategies.

2. ADVANCED REACTOR SYSTEMS PURSUED IN EUROPE

Fast reactors are the basis for a long term sustainable nuclear energy policy, as has been recognized at the European level within the European Strategic Energy Technology Plan. On the initiative of the Sustainable Nuclear Energy Technology Platform (www.snetp.eu), a European Strategic Energy Technology Plan industrial initiative named the European Sustainable Nuclear Energy Industrial Initiative has been launched. It gathers the main research and industry players in the field. Three ways are followed (all three of which are in conformance with the roadmap of the Generation IV International Forum), namely those cooled with sodium (SFR), lead or lead–bismuth (LFR) and gas (GFR). The EURATOM Framework Programmes (cordis.europa.eu), presently FP7 2012–2013 and the future HORIZON 2020 (from 2014 to 2018) support these initiatives through several projects (e.g. ESFR, LEADER, GoFastR, ALLIANCE, F-BRIDGE ARCHER, GETMAT, FAIRFUELS, PELGRIMM, JASMIN), which are more than ever focusing on safety aspects upon the request of the European Council. Fuels, materials and basic (actinides, nuclear data) supporting research programmes are also carried out, within which the Joint Research Centre (JRC) plays an important role through its institutional programme and partnerships in international projects.

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 license the advanced fuels. This will be one of the major roles of MYRRHA in particular. 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.

In addition to Pu which will be recycled in fast reactors, a further reduction of high level waste radiotoxicity and 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 reactor. Today, three ways to achieve transmutation of the minor actinides are considered and described. The reprocessing technologies and the fuel or target types have to be adapted to the selected option.

Considering transmutation in ADS, the main EURATOM effort will be placed on the safety assessment studies related to the MYRRHA demonstration projects by SCK/CEN. The project named MAXSIMA will include accidental events studies with a focus on transients potentially leading to pin failures. The project also incorporates validation experiments with mixed oxide (MOX) fuel for safety computer codes. Fuel–coolant–clad chemical interactions will be studied up to 1700°C.

In all options, fuel steady state and transient experiments will need to be carried out to a large extent, which requires resources and is a long term challenge. That is the reason why in the demonstration projects now under the 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.

3. ADVANCED FUEL CONCEPTS

Minor actinide transmutation can be performed in three main ways:

(i) Homogeneous recycling requires the integration of all minor actinides in every fuel pin in the reactor core. Thus, the minor actinide content could vary from just less than 1%, in the case of an equilibrium fuel cycle, up to several per cent, should the fast reactor fleet be deployed to transmute minor actinides from other reactor parks. Owing to the small content of minor actinides, minimal perturbations of the fuel performance under irradiation, or of its behaviour in recycling industrial plants, should be expected, but nevertheless remain a challenge and need further safety assessment. The homogeneous recycling mode also provides proliferation resistance due to the presence of minor actinides in the fuel. It has the disadvantage, however, that the entire fuel fabrication facility would require heavy shielding and process automation.

(ii) In contrast, heterogeneous reactor recycling of minor actinides can be achieved by incorporation of much higher quantities of minor actinide (~20%) in dedicated fuel assemblies, most likely in the blanket regions (i.e. UO_2 should be the matrix) of the reactor core. In this way, standard MOX driver fuel can be used in the central core regions, and standard fabrication plants can be used to produce it. The behaviour of minor actinide blankets (no fissile component in the fresh fuel) in-pile is relatively unknown, as are their thermophysical and thermochemical properties. Given the high minor actinide content in targets, the production capacity of dedicated minor actinide target fabrication facilities can be smaller than those for the driver fuel.

(iii) Finally, the ADS route has the advantage of enabling reaching the highest concentration of minor actinides in the core owing to its neutron physics characteristics. The fuels will be composed of Pu (including second generation Pu) and minor actinides in high concentration (up to 40%), embedded in inert matrices (references are Mo and MgO). The price to pay will be the development of a new type of reactor (linked to an accelerator) and very innovative fuel.

Non-oxide compounds such as metal alloys, nitrides and carbides are also considered as possible nuclear fuels for the new generations of nuclear power reactors, because of their high fissile density, good compatibility with sodium and low smear density, permitting very high burnups. More details on fuel options can be found in a dedicated presentation on recent advances in fuel for fast reactors given by Somers at this conference [3].

3.1. Safety relevant properties of advanced fuels

In general, the fuel options described above are characterized by high concentrations of plutonium and minor actinides, which have a significant impact on the physical, chemical and radiation properties of the fuel, affecting their production process, handling and irradiation behaviour. Fuel safety aspects of the Gen IV GFR, SFR and LFR systems are studied in a comprehensive set of investigations covering basic fuel properties, fuel coolant and cladding interactions, and irradiation behaviour with the final goal of establishing safety limits for production and in-pile performance of the advanced fuels.

Avoiding fuel melting is a major safety issue for nuclear fuels. Therefore, experimental efforts concentrate particularly on investigating the thermodynamic properties (free energy, thermal conductivity, melting point, helium and vaporization behaviour) and phase diagrams of advanced fuels in the major regions of interest. Since uranium and plutonium dioxides are the major constituents of oxide fuels, the UO_2-PuO_2 pseudo-binary system is of paramount importance. In view of recent new data for the melting point of PuO_2 [4–6], the high temperature phase diagram of UO_2-PuO_2 has been reinvestigated by means of more advanced experimental methods. In this context, the melting behaviour of MOX samples containing 75, 80 and 90mol% of PuO_2 was studied by laser heating under a controlled atmosphere [7]. The solidus and liquidus points measured (see Fig. 1, red data points) for these compositions were found to be much higher than those proposed by earlier researchers, based on more traditional thermal analysis methods. In line with the new melting point of pure PuO_2 (higher than 3000 K), these experimental data suggest that the solidus and liquidus lines for these systems should meet at a common minimum at a composition probably lying between 50 and 70mol% PuO_2. Further studies are needed on this system to establish the minimum temperature and the impact of oxygen potential and redistribution on the ternary U-Pu-O phase diagram.

The integration of the newly obtained information into the development of models to predict the performance of these fuels is a necessary step in the determination of the in-pile operational limits of these advanced fuels for safety authorities.

In this context, fuel rod performance codes play an important role, among them the TRANSURANUS code developed at JRC-ITU. This was originally developed for fast reactor performance analyses during the 1970s. Afterwards, however, almost all the efforts were dedicated to LWR modelling so that nowadays the code is considered as a best estimate tool for LWRs as proven by the extensive validation and assessment carried out at the JRC-ITU. However, the fast reactor version of the code has not benefitted from the same experience. Therefore, there is a need for an upgrade, extension and assessment

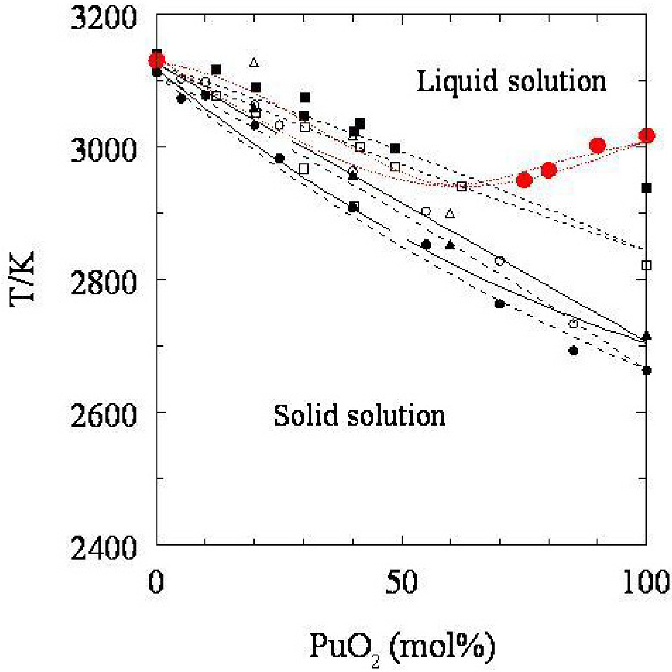

FIG. 1. The solidus–liquidus lines in the UO_2-PuO_2 phase diagram (red points: new data measured by laser heating) [7].

of the models for fast neutron reactors. Various developments have been carried out in the frame of EU funded projects such as F-BRIDGE, GOFASTR and PELGRIMM. They deal with the burnup and helium production model, the radial redistribution model for plutonium, oxygen and minor actinides, the normal grain growth model, and a new helium release model. All these developments make use of a so-called multi-scale approach in which experimental data and detailed simulation techniques (e.g. finite elements, classical dynamics and ab initio computations) are used.

As a concrete example, the plutonium redistribution model (PUREDI) calculates the steady state and transient radial plutonium concentration as a function of the radial temperature profile and the time for MOX fuels. In the context of developing a TRANSURANUS version for Gen-IV fast reactor fuels, PUREDI has recently been modified and extended to include the effects of the local power density profile and the oxygen-to-metal ratio on plutonium transport. The model has been extensively verified by means of specific numerical tests, and after incorporation in the TRANSURANUS fuel performance code, has been assessed on the basis of post-irradiation examination (PIE) of the SUPERFACT experiment, showing a good agreement with the experimental data [8].

3.2. Irradiation behaviour of transuranium fuels

Irradiation experiments performed with different types of minor actinide containing fuels and irradiation targets are summarized in Table 1 below.

TABLE 1. SUMMARY OF THE IRRADIATION EXPERIMENTS WITH MINOR ACTINIDE FUEL

Experiment name	Reactor	Fuel type	Materials	Status
FACT		MOX		Completed
SUPERFACT	Phénix	MOX	$(U, Pu, MA)O_2$ $(U, MA)O_2$	Completed
TRABANT-1	HFR	MOX	$(U, Pu, Np)O_2$	Completed
Am-1	Joyo	MOX	$(U, Pu, Am)O_2$	Completed
X501	EBR-II	Metal	(U, Pu, Zr, Np, Am)	Completed
Metaphix	Phénix	Metal	(U, Pu, Zr, Np, Am, Cm)	PIE ongoing
EFTTRA-T4	HFR	Inert matrix	$MgAl_2O_4 + AmO_2$	Completed
ECRIX	Phénix	Inert matrix	$MgO + AmO_2$	PIE ongoing
CAMIX-COCHIX	Phénix	Inert matrix	$MgO + (Zr, Y, Am)O_2$ $(Zr, Y, Am)O_2$	PIE to be started
HELIOS	HFR	Inert matrix	$Mo + (Pu, Am)O_2$ $Mo + (Zr, Am)O_2$	PIE started
FUTURIX-FTA	Phénix	Various	$Mo + (Pu, Am)O_2$ $Mo + (Zr, Pu, Am)O_2$ $MgO + (Pu, Am)O_2$ (U, Pu, Zr, Am, Np) (Pu, Am, Zr) (Zr, Pu, Am)N (U, Pu, Np, Am)N	PIE to be started

Extensive research on various dispersion type transmutation targets has been performed by the EFTTRA group [9] and a variety of matrices was studied in irradiation experiments in the High Flux Reactor (Petten, Netherlands) and Phénix (France).

MOX fuels containing minor actinides were irradiated as part of the SUPERFACT experiment in the Phénix reactor (France) during the 1990s (see Fig. 2). In this experiment, the irradiation behaviour during normal conditions of minor actinide fuels with four different concentrations, were studied and compared to standard Phénix fuel. All fuels were produced by the sol-gel technique, resulting in a homogeneous distribution of the minor actinides in the matrix. The linear power was between 17 and 28 kW/m for high actinide content pins, between 38 and 32 kW/m for low actinide content pins, and 43 and 37 kW/m for the accompanying standard Phénix pins.

Fast reactor metal fuels containing minor actinides (Np, Am, Cm) and rare earths (Y, Ce, Nd, Gd) were irradiated in the fast reactor Phénix in the METAPHIX project. METAPHIX is a collaboration programme between the Central Research Institute of Electric Power Industry (CRIEPI, Japan), the JRC-ITU and the CEA (France). Its goal is the study of the safety of a closed cycle based on U-Pu-Zr metal alloy fuel containing minor actinide and rare earths. Nine Na bonded fuel test pins were prepared at ITU and loaded with U-Pu-Zr-based alloy ingots. Three ingots contained minor actinides; two of them, in addition to minor actinides, contained rare earths, reproducing the output of pyrometallurgical reprocessing of LWR spent fuel [10]. The pins were irradiated in the Phénix reactor with the support of the CEA. The three assemblies (METAPHIX-1, 2 and 3) achieved burnups of ~2.5at.%, ~7.0at.% and ~10.0at.%, respectively [11].

PIE of the SUPERFACT fuels showed good in-pile performance not very different from the standard MOX fuel of Phénix. The non-destructive examinations of the four fuel types did not show any anomaly in their behaviour. The homogeneously recycled fuel operated at high linear power shows the typical central hole formation which is well known in standard MOX fuel. Fission gas release rates (60–80% of that produced) were in good agreement with those of standard MOX fuels deployed in the same assembly, even for fuels with a high concentration of minor actinides. The high concentration of americium in a $(U_{0.60}Np_{0.20}Am_{0.20})O_2$ fuel led to significant increases of the fuel column length and diametrical deformation of the cladding, probably due to mechanical interaction between the oxide fuel and the cladding. Furthermore, this pin showed high helium production which may have contributed to the fuel swelling. The efficiency of transmutation in all fuels was about 30%, which clearly shows that multiple actinide recycling will be required.

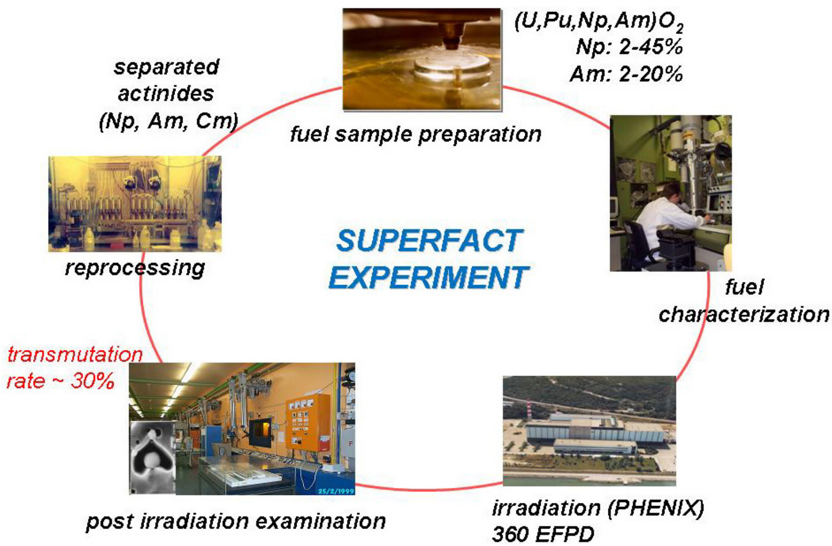

FIG. 2. Closing of the fuel cycle as demonstrated by the SUPERFACT experiment.

PIE of the METAPHIX samples has been carried out in JRC-ITU. Non-destructive examination results [12] provided information on pin and fuel swelling and relocation of bonding sodium to the plenum. Destructive PIE so far has been performed on the low and medium burnup fuels [13]. Analysis of the fission gas released to the plenum during irradiation was performed [14] showing that the behaviour of the fuel containing minor actinides up to ~7at.% burnup was in line with U-Pu-Zr fuel irradiated in the EBR-II reactor [15]. Ongoing work is focused on possible effects associated with the generation and release of helium from minor actinide-containing fuel.

Optical and electron scanning microscopy has been performed to characterize morphology and phase distribution in the irradiated fuel. Figure 3 shows the optical microscopy image of a cross-section of fuel with 5% minor actinide and 5% rare earths irradiated to a burnup of 7at.%. A dense phase characterizes the central region of the fuel and this is surrounded by an intermediate porous phase. Near the radial periphery, higher porosity and some cavities are evident. Electron microprobe analysis is ongoing to gather detailed information concerning fission products relocation and characterize the phases which formed during the irradiation.

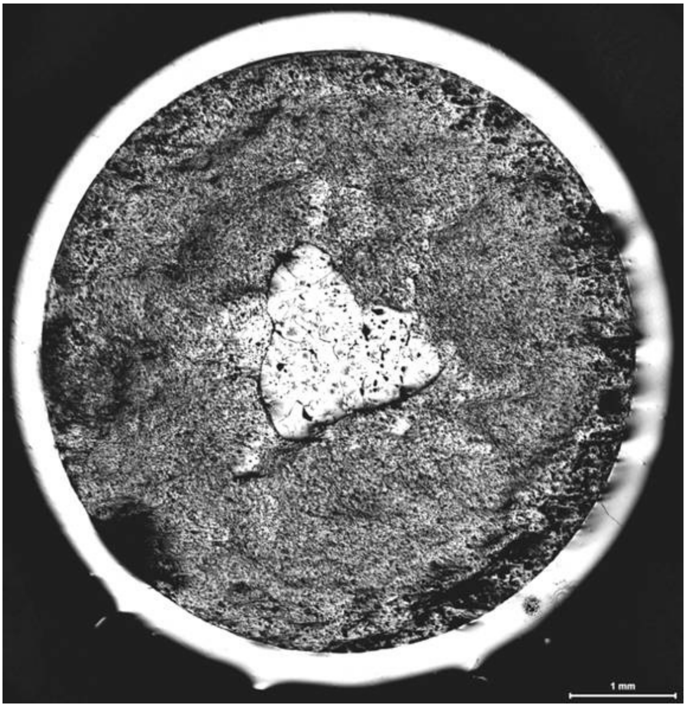

FIG. 3. Cross-section optical microscopy image of U-Pu-Zr fuel containing 5at.% minor actinide and 5at.% rare earths irradiated to a burnup of 7at.%.

4. SEPARATION STRATEGIES

Two types of processes can be applied to the separation of long lived radionuclides: hydrochemical (wet) and pyrochemical (dry) processes. Both have advantages and disadvantages and should be applied in a complementary way. The decision on the partitioning process to be applied should depend on the boundary conditions, such as the type of fuel material to be treated. In any case, an efficient and selective recovery of the key elements from the spent nuclear fuel is absolutely essential for a successful sustainable fuel cycle. This necessitates that Am and Cm can be selectively separated from lanthanide fission products, certainly the most difficult and challenging task in advanced recycling of spent nuclear fuel owing to the very similar chemical behaviour of trivalent elements.

In a so-called double strata concept, the well established industrial reprocessing of commercial LWR fuel with recycling of U and Pu based on PUREX extraction could be combined in the first stratum with an advanced aqueous partitioning scheme also based on liquid–liquid extraction to separate the minor actinides.

For the separation and recovery of Am and Cm from the PUREX highly active raffinate, several extraction processes have been developed worldwide [16, 17]. One of the most promising is the French DIAMEX-SANEX process, further developed in European collaborations. Diamides are used to co-extract trivalent lanthanides with Cm and Am and BTBP molecules are used to separate actinides from lanthanides. Besides the excellent extraction capabilities, the adaptation to the PUREX process is straightforward and the solvent is completely combustible and therefore yields no solid residue. The combined PUREX-DIAMEX-SANEX process has been successfully demonstrated at JRC-ITU using actual fuels. The challenge now is to optimize the developed processes in terms of their transferability towards industrial process maturity. This will only be possible in the form of a large international project, and this is being addressed in the current EU project ACSEPT, which will be followed by the new collaborative project SACSESS.

In Ref. [18], it is suggested that the implementation of an innovative SANEX process would result in a simplification of the processes required for trivalent actinide separation. A combination of An(III) + Ln(III) co-extraction by a diamide or diglycolamide solvent (e.g. TODGA) and subsequent selective stripping of An(III) by a strong hydrophilic complexing agent (e.g. SO_3-Ph-BTP) seems to be a promising process for the separation of Am(III) and Cm(III) from a PUREX raffinate solution. All these innovative separation processes are based on solvent extraction, involving new extracting or complexing organic molecules and new diluents. The mechanistic understanding of the chemical and physical reactions involved (thermodynamic and kinetics) still needs to be improved further to develop multiscale models to be used in simulation codes.

Pyrochemical separation techniques [17, 19] are relevant alternative options for the advanced separations. They could be the preferred method for advanced oxide fuels (mixed transuranium, inert matrix or composite), metal fuels or nitride fuels, because of a limited solubility of some of the fuel materials in acidic aqueous solutions. Other potential advantages of the pyrochemical approach to recycling advanced fuels, in comparison to hydrochemical techniques, are higher compactness of equipment and the possibility to form an integrated system between the irradiation and the reprocessing facilities, thus reducing considerably transport of nuclear materials. In addition, the radiation stability of the salt in the pyrochemical process as compared to the organic solvent in the hydrochemical process offers an important advantage when dealing with highly active spent minor actinide fuel. The fuels will be irradiated to a very high burnup and shorter cooling times would certainly reduce the storage cost.

5. PROLIFERATION RESISTANCE AND PHYSICAL PROTECTION

In addition to the features affecting sustainability, safety, reliability and economics, all Generation IV International Forum nuclear energy systems (NES) should exhibit advanced proliferation resistance and physical protection features with respect to the existing systems.

Nuclear installations worldwide are controlled under the IAEA safeguards system. The safeguards system comprises an extensive set of technical measures by which the IAEA independently verifies the correctness and the completeness of the declarations made by States about their nuclear material and activities. The IAEA defines proliferation resistance as "… that characteristic of an NES that impedes the diversion or undeclared production of nuclear material or misuse of technology by the Host State seeking to acquire nuclear weapons or other nuclear explosive devices" [20]. Proliferation resistance depends on intrinsic features of the facility and on extrinsic measures. Intrinsic features result from technical design features, including those facilitating the implementation of safeguards measures, while extrinsic measures results from state decisions and undertakings. For this reason, under the auspices of the Generation IV International Forum, the Proliferation Resistance and Physical Protection Working Group (PR&PP WG) has developed a methodology for the PR&PP evaluation of Generation IV International Forum NES [21].

Here, the inclusion of minor actinides in the fresh fuel increases the radiological barrier of one of the most attractive targets for a potential nuclear proliferator and is hence generally considered to contribute to proliferation resistance. On the other hand, the inclusion of minor actinides to fresh fuel may pose safeguards problems such as inspection accessibility. In addition, it can also make the isotopic identification of the fresh fuel for safeguards verification purposes more complex [22]. Moreover, the increased technical difficulty related to handling the minor actinide-bearing fresh fuel might have a non-irrelevant impact on the routine operators' activities. Evaluating the proliferation resistance of a facility or of a whole fuel cycle is thus a complex matter implying structured methodologies, such as the one developed by the Generation IV International Forum's PR&PP WG. While designing new facilities and advanced fuel cycles, it is important that safeguards are considered from the early design phases according to the safeguards by design concept [23]; this will also contribute to fostering a safeguards and security culture and to creating synergies with safety.

6. 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 [24]. Concerns have been raised that nuclear education and training is not at the level where it should be, as summarized in the OECD/NEA report [25]. 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" [26].

In response to this, the European Nuclear Education Network has been established and the Euratom FPs have given priority to strengthening education and training either by stimulating an editing component in funded research projects or through horizontal activities such as research infrastructures and human resources, training and mobility (e.g. TALISMAN/ACTINET, CINCH (see cordis.europa.eu)). Furthermore, several universities in the European Union 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 generally lack 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.

As a result of these specific infrastructure needs, nuclear education and training has not been the domain of the academic institutions alone, in contrast to other fields of science and technology. Whereas nuclear reactor technology is generally well covered at universities, the knowledge and capabilities of handling nuclear materials is traditionally concentrated in national and international research facilities that have the appropriate infrastructure, ensuring the high level of safety and security required. As a result, 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 Joint Research Centre has taken the initiative to establish the European Nuclear Safety & Security School centred on the facilities of the JRC-ITU in order to make its nuclear research facilities more accessible for training and education programmes.

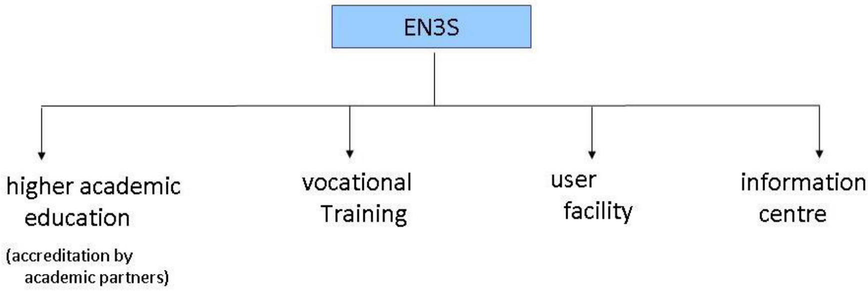

FIG. 4. Components of the European Nuclear Safety & Security School.

In collaboration with relevant partners from Europe, the European Nuclear Safety & Security School will be based on educational and training tracks that make use of its unique facilities and knowledge in the fields of nuclear security and safety, nuclear materials, nuclear data and actinide science. In particular, students will have the possibility of gaining hands-on experience and participating in cutting edge research (e.g. within the frame of European projects such as GENTLE or TALISMAN). In parallel, the JRC-ITU and its academic partners will join efforts to offer vocational training for (young) researchers and engineers working in, among others, industry, consultancy companies, public authorities, regulatory agencies and technical safety organizations to guarantee their continuous professional development. Figure 4 shows the four main directions of the European Nuclear Safety & Security School that have been identified.

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PROGRESS IN PYROPROCESSING TECHNOLOGY AT KAERI

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Abstract

There were 23 nuclear power plants in the Republic of Korea in 2012. The spent nuclear fuel management from nuclear power plants is an issue for sustaining nuclear power. Pyroprocessing is one of the options with which to manage the spent nuclear fuel, which has obvious benefits over the other alternatives. Individual units of pyroprocessing have been studied for decades to facilitate innovative technologies and related optimization studies. The research activity for integrating the unit processes was required for the verification of overall unit process performance and this led to the PRIDE facility construction. The purposes of the PRIDE facility is to facilitate integration of tests of unit processes, application of safeguards concept, optimization of processes, and provision of scale-up information and a training pyroprocess facility.

1. INTRODUCTION

As of August 2012, the Republic of Korea has 23 nuclear power plants accounting for about 30% of total electricity generation. Consequently, about 12 000 t of spent fuel in total is stored in the pool at plant sites. The management of spent fuel is now a national issue, since the capacity of the pools at the sites will be saturated in the near future. The national policy for the ultimate management of spent nuclear fuel has not yet been determined. Alternatively, interim storage is now a consideration.

Interim storage, however, cannot ultimately resolve the spent fuel issue. Besides interim storage, other options for the ultimate solution of spent fuel management are needed to be studied; one is direct disposal and the other is disposal after treatment based on the closed fuel cycle concept.

The benefit of closed fuel cycle is obvious. The amount of waste can be reduced remarkably by recovery of useful uranium, which constitutes a major part of spent fuel. The repository area can be reduced by removal of heat emission elements. Burning of long lived elements in a fast reactor also decreases the period of radioactive waste returning to being a natural level of radioactivity.

These are attractive factors to a country such as the Republic of Korea where there is no abundant land area for disposal.

The Korea Atomic Energy Research Institute (KAERI) has pursued pyroprocessing technology development to obtain the benefits by treatment of spent fuel since 1997. The advantages of pyroprocessing compared with other conventional processes have been published elsewhere [1, 2]. This paper outlines the achievements of pyroprocessing technology developments and the PRIDE (PyRoprocessing Integrated inactive Demonstration) facility.

2. PYROPROCESSING

2.1. Head end process

The spent fuel conveyed from the pool is decladded for treatment. The UO_2 is voloxidized at high temperature to remove volatile elements such as Cs, Tc, I, H-3, Kr, Xe, etc. It was deteremined experimentally that 98% of Cs and 100% of Kr, Xe, H-3, C-14 and I-129 could be removed. The removed volatile elements should be trapped by a filtering system. It was demonstrated that released Cs was completely trapped by a fly ash filter [3].

Cs is trapped on a fly ash filter at around 1000°C; Tc is trapped on a calcium filter at about 800°C, and I is trapped on a AgX at about 150°C (see Fig. 1). The H-3 is trapped on a molecular sieve after conversion to HTO by using CuO and Kr-85 is cryogenically condensed and then trapped on solid adsorbent.

FIG. 1(a). Fly ash filter.

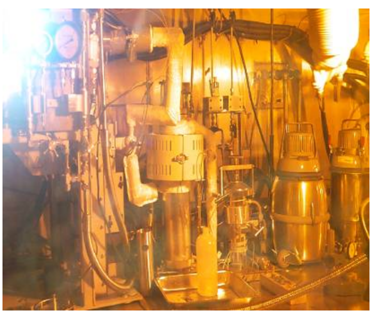


FIG. 1(b). Off-gas trapping system.

2.2. Electrolytic reduction

The oxide spent fuel from the head end process is transported to the electrolytic reduction process in order to convert the oxide form to metal. Since this process handles the largest amount of material within the whole pyroprocess, compared with the other unit processes, then scale-up technology is important. The rate determining step was evaluated to estimate the process efficiency, yielding an optimized anode/cathode area ratio. Platinum is used as an anode. The alternative anode needs to be developed to consider the economic aspect. In order to minimize the Pt dissolution, optimum Li_2O concentration was obtained experimentally [4]. More than a 99% reduction was accomplished when uranium oxide was tested. Some rare earth elements (REE) could not be completely reduced, being transporting to the following process.

Reduced metal from the electrolytic reduction system contains salt. This salt is recovered at the distillation system. More than 99.9% of salt could be recovered using a specially designed distillation system. Figure 2 illustrates the distillation system.

2.3. Electrorefining

The purpose of the electrorefining process is to recover most of the uranium which constitues more than 92% in spent fuel. When the uranium is depleted in the salt, the recovery potential increases, for example, plutonium will co-deposit with uranium at the cathode. The requirement of electrorefining product is pure uranium in order for the product not to be treated as waste. This implies there is in the operation condition an upper limit of Pu/U ratio in the salt. If U and Np are to be completely removed for the other purpose, then chlorine gas evolves at the anode. This indicates obviously that electrorefining process is not able to

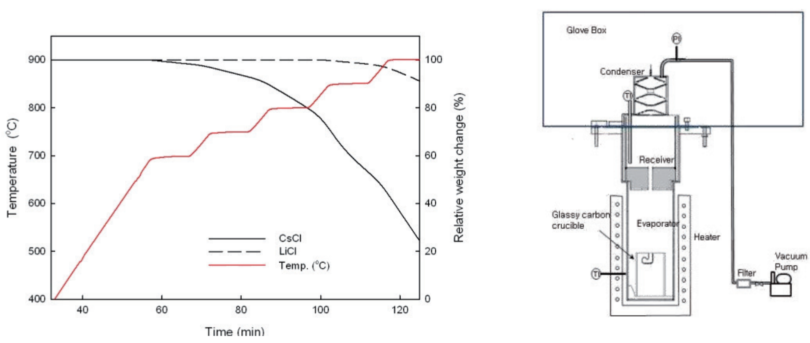


FIG. 2. Salt evaporation experiment.

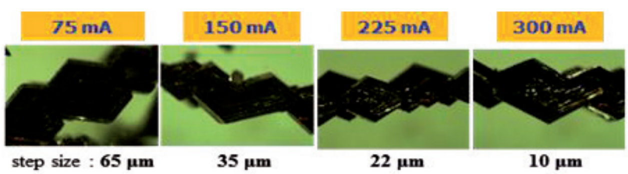
covertly be used for the other purpose. Needless to say it is strong evidence that pyroprocessing has an intrinsic high proliferation resistance. Recovered pure uranium contains salt. This mixture is conveyed to the distillation process to recover the salt. Distilled uranium is cast to produce an ingot. Figure 3 shows that the salt content in the product depends on applied current [5].

Partly unreduced REEs react with UCl_3 in the salt to give a rare earth chloride and uranium oxide. Uranium oxide may float on the salt and will be collected sometimes and fed back into electrolytic reduction system.

2.4. Electrowinning

The salt transported from electrorefining contains residual U, TRU and REEs. These elements are recovered altogether by a liquid cadmium cathode. The Gibbs free energy of uranium and the other elements differ slightly from each other. However, if a liquid cadmium cathode is applied to recover the elements then the differences between the elements become negligible due to their different activity coefficients. This product is impure, and thereby enhances proliferation resistance. The product mixture in the cadmium is distilled to recover cadmium.

The design goal of electrowining is to recover 99.9% of TRU. The other design goal is associated with the TRU/REE ratio, since the product should satisfy the feed material condition for fast reactors. Therefore, the process needs to recover TRU as much as possible with limited REE content. This requires the need for a residual actinide recovery system [6]. In the residual actinide recovery system, all TRU and REEs are co-deposited together in the liquid Cd initially, then $CdCl_2$ is introduced to the system so that $REECl_3$ can be re-dissolved to the salt. Consequently, the product is TRU mixed with adjusted REE in Cd. The product is again distilled to recover TRU and the REE mixture. Figure 4 shows the residual actinide recovery process' experimental results.



Current	Total weight	Salt weight	Salt ratio
75 mA	2.00	0.62	31%
150 mA	2.25	0.87	37%
225 mA	2.40	1.01	42%
300 mA	2.83	1.43	50%

FIG. 3. Dendrite characteristics with applied current.

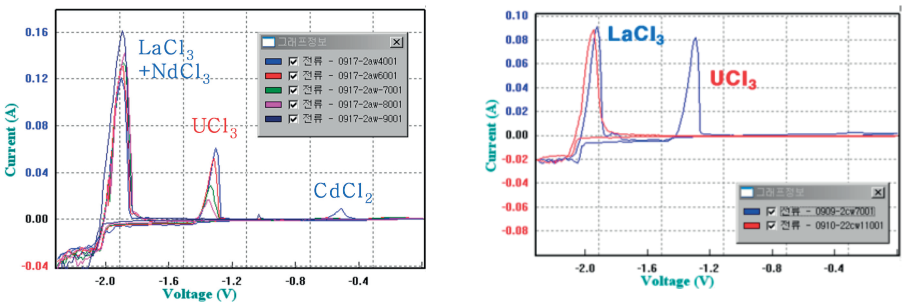


FIG. 4. Residual actinide recovery process experiments.

2.5. Waste salt treatment

In the pyroprocessing, there are two waste salts: LiCl from the electrolytic oxidation system and LiCl+KCl eutectic salt from the electrowinning system [7, 8]. The LiCl contains Sr, which is one of the major heat emitting elements. The crystallization method is applied to recover LiCl. The cold plate is immersed in the contaminated hot LiCl salt, then pure LiCl is solidified at the cold surface of the plate. Most parts of the LiCl salt can be recovered by this way. The residuals contain LiCl and contaminants. The final waste is mixed with a solidifying agent to form final solid waste.

For the recovery of eutectic salt from the electrowinning process which contains REEs and FS, a precipitation method is applied. Air or oxygen is mixed with contaminated salt, yielding an oxide of contaminants. After settlement, the upper layer, which is pure eutectic salt, is extracted and returned to the system. The lower part is distilled to recover the residual salt. The waste is mixed with a solidifying agent to give the final waste form. Figure 5 illustrates the waste form flow.

2.6. PRIDE

PRIDE (see Fig. 6) aims at testing integrated unit processes. Pyroprocessing technology has been studied on a laboratory scale. The unit processes were studied separately. As the technology was being developed, the test for overall unit process performance was needed to verify overall function of pyroprocessing.

The cell structure on the 2nd floor is 40 m long, 4.8 m wide and 6.4 m high. The atmosphere filled with Ar is controlled to satisfy the condition of oxygen and moisture concentrations of less than 50 ppm each. One bridge transported duel arm servo manipulator [9] capable of handling 25 kg with each arm, one overhead crane capable of lifting 3 t and one hoist capable of lifting 1 t, were

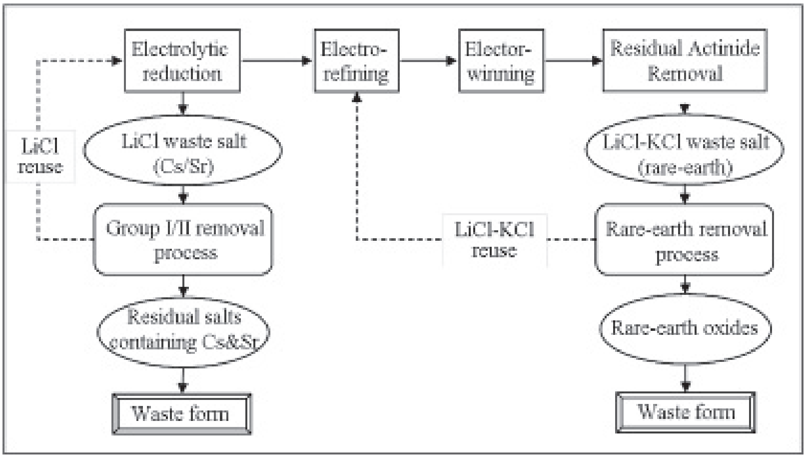


FIG. 5. Waste salt treatment flow.

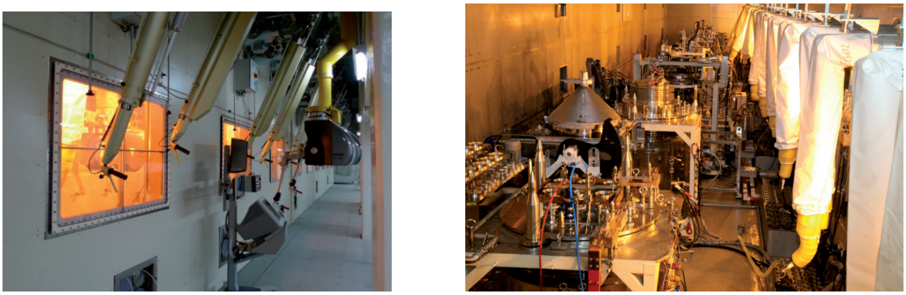


FIG. 6. PRIDE view.

installed on the ceiling. Seventeen sets of MSM capable of 15 kg at 17 windows were installed for handling the equipment. The material is transported into the cell through a large transfer lock, a small transfer lock, and two gravity tubes.

All equipment items were tested to check their remote operability before fabrication. When the equipment items were designed, they were tested by 3-D dynamic model computer simulation. The results were used to improve design and to fabricate the equipment. After fabrication, the equipment items were tested again with a mock-up remote operability test system which has one window, one set of MSM, one bridge transported duel arm servo manipulator and one overhead crane. Minor corrections for equipment were made through these mock-up test results.

PRIDE will test the equipment operation, interconnection between equipment, material flow measurement and integral process performance. It will provide beneficial information pertaining to scale-up design, operation and the safeguards concept [10].

3. CONCLUSION

Spent fuel treatment technology is certainly attractive to many countries which have spent fuel management issues, since it benefits from a remarkable reduction in the high level waste repository area. A small sized country with many nuclear power plants, such as the Republic of Korea, will be interested in a spent fuel management option that reduces the repository area. Pyroprocessing is an adequate option to satisfy this requirement.

Unit pyroprocessing has been developed at KAERI for decades, and the PRIDE facility was constructed with the aim of testing the integrity of unit processes. It will provide scale-up related information and safeguards information as well. Through experience gained through the PRIDE facility operation, pyroprocessing will be realized in the near future.

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THE UNITED STATES DEPARTMENT OF ENERGY'S FUEL CYCLE TECHNOLOGIES PROGRAMME

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Abstract

As with all nations, the United States of America has a significant interest in ensuring the affordable, secure and sustainable supply of energy to power its economy and homes. The Office of Nuclear Energy within the US Department of Energy is conducting research and development to support this overarching goal to ensure that nuclear energy remains a key component of the US portfolio of energy technologies. In 2010, the Office of Nuclear Energy adopted four objectives to guide its research and development (R&D): (i) develop technologies that can improve the reliability, sustain the safety, and extend the life of current reactors; (ii) develop improvements in the affordability of new reactors to help meet energy security and climate change goals; (iii) develop sustainable nuclear fuel cycles; and (iv) understand and minimize risks of nuclear proliferation. In support of the third objective, the Office of Fuel Cycle Technologies is conducting a comprehensive, systems level evaluation and screening of fuel cycle options. The results of this effort will help identify which fuel cycle options have the potential for substantial beneficial improvements in performance compared with the current fuel cycle and to support the prioritization of R&D investments.

1. UNITED STATES DEPARTMENT OF ENERGY OFFICE OF NUCLEAR ENERGY FUEL CYCLE TECHNOLOGIES PROGRAMME

Powerful imperatives drive the continued needs for nuclear power, among them are escalating energy demands, global warming, and volatility in the fossil fuel supply. As the only large scale source of nearly greenhouse gas-free energy, nuclear power is an essential part of the United States of America's energy base, generating about 20 per cent of our nation's electricity and almost 70 per cent of the clean energy. Despite its prominent role, nuclear energy's use presents challenges, such as mounting stockpiles of used nuclear fuel and high level waste and a potential for proliferation of nuclear materials. The March 2011 events at the Fukushima Daiichi nuclear power plant underscored the urgency behind enhancing accident tolerance of the existing reactor fleet. The USA must address these challenges in order to meet its goals for energy, environmental and economic security.

Nuclear power is making major contributions towards meeting the nation's current and future energy demands. The USA must continue to ensure improvements and access to nuclear technology to meet its economic, environmental and energy security goals. The USA relies on nuclear energy because it provides a consistent, reliable and stable source of base load electricity with an excellent safety record in the USA. In order to continue or expand the role of nuclear power in its long term energy platform, the USA must:

- Continually improve the safety and security of nuclear energy and its associated technologies;
- Develop solutions for the transportation, storage and long term disposal of used nuclear fuel and associated wastes;
- Enhance the resilience of nuclear plants and used nuclear fuel in storage to extreme events and beyond design basis accidents such as at that which affected the Fukushima Daiichi plant;
- Improve the long term sustainability of nuclear energy.

To address these high level goals, the Fuel Cycle Technologies (FCT) programme of the US Department of Energy (DOE) Office of Nuclear Energy (NE) is charged with identifying promising sustainable fuel cycles and developing strategies for effective disposition of used fuel and high level nuclear waste, enabling US policymakers to make informed decisions about these critical issues.

The FCT programme has taken a dual focus approach to address near and long term research and development (R&D) challenges to expand the role of nuclear power in the US energy platform. In the near term, nuclear power will remain a critical part of the US energy mix and the challenges are to maximize the performance and safety of the current fleet. In the long term, the challenge is to develop systems that enable improved sustainability of nuclear energy. Concurrent to both R&D paths are the ongoing concerns of managing the risk of proliferation of nuclear materials. Thus, the programme has defined a dual focus strategy to simultaneously support the use of nuclear power today and investigate

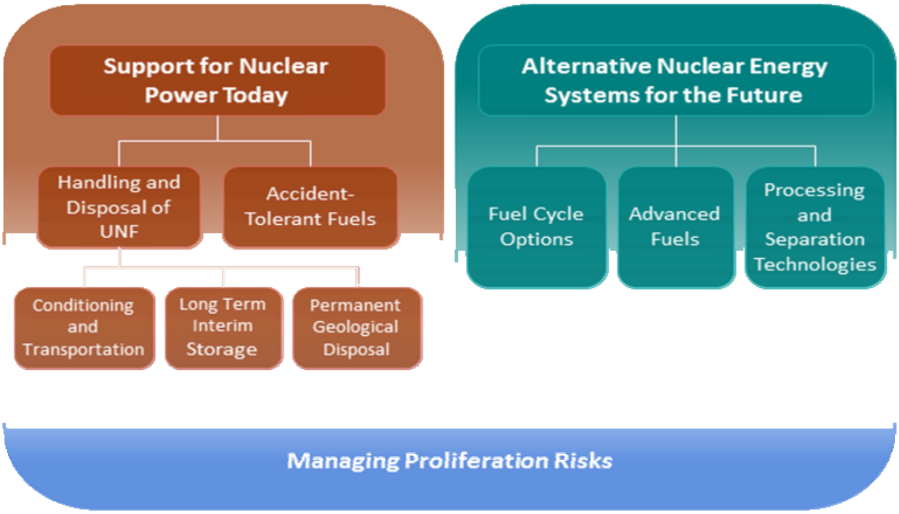

FIG. 1. FCT dual focus strategy.

nuclear energy system options for the future, while managing proliferation risk (Fig. 1).

In addition to incorporating a dual focus approach, the FCT has structured five technical areas that play an important role in meeting near and long term challenges. To effectively accomplish its mission, the FCT focuses on R&D in five technical areas that span the entire nuclear fuel cycle:

(i) **The fuel cycle options area** is developing management processes and tools and performing integrated fuel cycle technical assessments to provide information that can be used to guide the selection of sustainable options.

(ii) **The advanced fuels area** is developing proliferation resistant, next generation metallic fuels for the recycling of transuranics, along with advanced accident tolerant fuel for current light water reactors.

(iii) **The separations, waste forms and fuel resources area** contributes to both a sustainable fuel cycle and improved waste management by effectively recovering transuranic elements from used nuclear fuel and seeking transformational breakthroughs in waste forms with greatly improved performance.

(iv) **The used fuel disposition area** is enabling the technology for storage, transportation and disposal of used nuclear fuel and wastes generated by existing and future nuclear fuel cycles.

(v) **The materials protection, accounting and control technologies area** is developing the technologies, monitoring tools and analysis techniques for next generation nuclear safeguards and security, minimizing the risks of proliferation.

Near term challenges (Fig. 2) exist related to technical recommendations for used fuel management, increasing the focus on nuclear fuels with enhanced accident tolerance, identifying sustainable fuel cycle options for further development and strengthening the technical and scientific basis for extended storage of used nuclear fuel and high level waste.

In the intermediate term, the programme will conduct science based, engineering driven research for sustainable fuel cycle options, conduct research to support extended storage of used nuclear fuel, and demonstrate and deploy the selected enhancement for accident tolerance.

In the long term, the programme will demonstrate specific fuel cycle technologies. The FCT will implement safe strategies for management of used nuclear fuel and high level waste, and deploy advanced nuclear systems for affordable, safe and secure nuclear generated electricity.

As part of its effort to continually improve its decision making, the DOE-NE developed and tested a method that will be applied to screening fuel cycle options.[1] That earlier work is currently being refined (Fig. 3) and the resulting comprehensive study on the evaluation and screening of fuel cycle options is planned to be completed in late 2013.

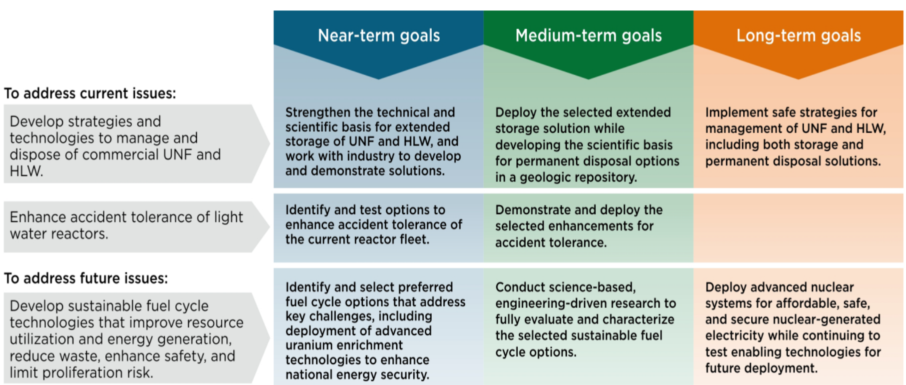

FIG. 2. Near to long term goals.

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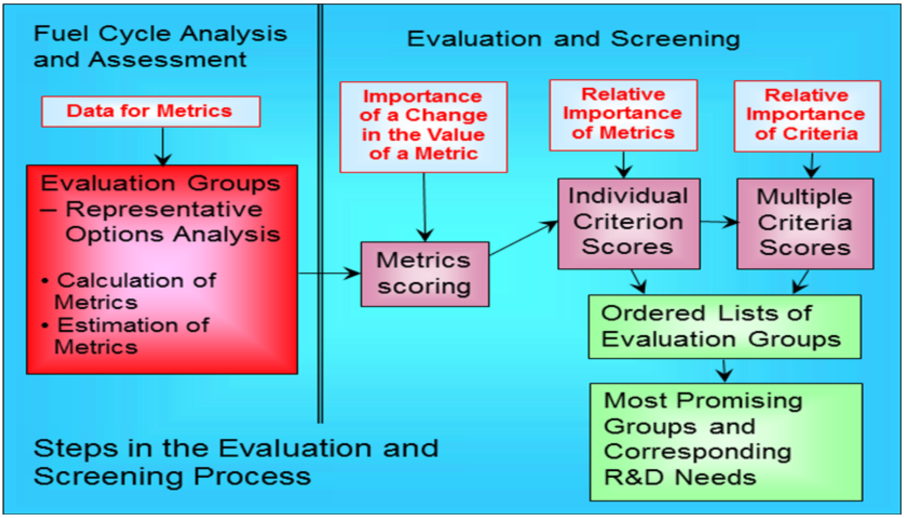

FIG. 3. Screening fuel cycle options diagram.

The purpose of the fuel cycle evaluation and screening is to evaluate the comprehensive range of nuclear fuel cycle options using a set of evaluation criteria and metrics, to identify fuel cycle options that have attractive characteristics with respect to nine high level evaluation criteria that reflect programme objectives and societal needs. The nine high level evaluation criteria are: (i) nuclear waste management, (ii) proliferation risk, (iii) nuclear material security risk, (iv) safety, (v) financial risk and economics, (vi) environmental impact, (vii) resource utilization, (viii) development and deployment risk and (ix) institutional issues. This information can be used for prioritization of R&D. The process should offer valuable input to decision makers while also giving them a clear understanding of how the results were obtained.

The evaluation and screening of nuclear energy systems consists of four major steps: (i) define what nuclear energy systems will be evaluated and screened, (ii) develop the metrics to evaluate any potential improvements, (iii) conduct the evaluation and screening and (iv) evaluate the results.

(i) Identification of systems for evaluation and screening

The set of nuclear energy systems that will be evaluated and screened has been created, starting with fundamental physical principles, and then verified, validated and completed using information from several sources, including prior studies, industry, universities, national laboratories and the public. For the purposes of screening fuel cycle options, those functions associated with mining and disposal (see Fig. 4) are represented generically without consideration of specific choices for the method of obtaining resources or the environments for disposal of wastes. An electronic database, called the Fuel Cycle Catalog, will house the descriptions and technical data for the comprehensive set of nuclear systems used for the evaluation and screening, including detailed information on system performance and any relevant technologies, as appropriate.

(ii) Development of evaluation metrics

Evaluation metrics are needed to conduct the evaluation and screening of nuclear energy systems. The evaluation metrics for each of the nine high level criteria mentioned above have been developed in cooperation with the FCT R&D programmes, other divisions of the DOE, industry and universities. For objectivity, it is desirable that the metrics are quantifiable to the extent possible, and the use of qualitative metrics requiring expert elicitation is used only when quantitative metrics are not possible.

(iii) Evaluation and screening

The evaluation and screening of the nuclear energy systems will be conducted by a team of experienced and internationally recognized scientists and engineers. An independent review team has also been engaged by DOE-NE to provide an independent review of all aspects of the evaluation and screening process and to ensure objectivity, transparency and the validity of the results. Several parameters are essential for conducting the evaluation and screening of fuel cycle options such as the weight and value assigned to a criterion or metric. Since these are highly subjective, sensitivity studies to explore the effects on the results will be conducted.

In Fig. 3, the relative importance of criteria refers to the weight assigned to each of the nine high level criteria in order to signify its importance relative to the other criteria. This is important given the recognition that different decision makers have different impressions of which criteria are more important than others

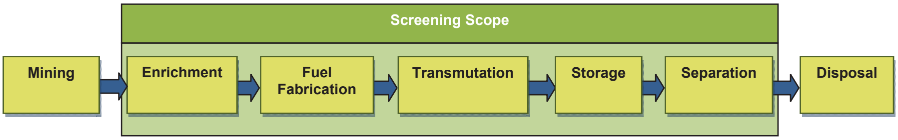

FIG. 4. Screening scope of fuel cycle options process.

in prioritizing R&D needs. For example, some may feel that waste management is more important than proliferation risk, but that both are important when a fuel cycle is evaluated. Each criterion for a fuel cycle is scored and then combined appropriately using the weighting factors to reflect the relative importance of each criterion to provide an overall score. As previously mentioned, sensitivity studies of different sets of weighting factors will be conducted as part of the evaluation and screening.

The final result of this process will be prioritized lists of fuel cycle options scored with respect to the evaluation criteria. This list will identify the most promising fuel cycle options that can be used to inform decisions on developing profitable R&D directions.

(iv) Evaluation of results

The results of the screening and evaluation will be used to answer the following questions:

- Which nuclear fuel cycle systems have the potential for substantial beneficial improvements in nuclear fuel cycle performance, and what aspects of these systems make these improvements possible?
- Where is DOE R&D investment needed to support the set of promising fuel cycle systems?
- What technical objectives and performance goals can be defined to guide programme research?

In conclusion, the evaluation and screening methodology provides a systematic, objective and transparent method for evaluating and categorizing nuclear energy systems according to their performance in meeting FCT programme objectives. This, in turn:

- Improves the programme's ability to clearly identify and prioritize research and R&D needs and better communicate the rationale for R&D directions, funding decisions and policy making;
- Enhances the ability of the programme to formulate and execute programme budgets;
- Allows the programme to more readily adapt to future policy changes with rapid determination of how any changes impact the prioritization of R&D for key technologies.

This methodology offers the prospect of facilitating dialogue among a variety of stakeholder groups interested in US energy policy and the future of nuclear energy in the USA by connecting the specific R&D directions to the evaluation criteria. The evaluation and screening activity is planned to be completed by the end of 2013 with a final report available to the public by May 2014.

EXPERIMENTAL TESTS, DATA AND ADVANCED SIMULATION

Chairpersons

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DOE–CEA BENCHMARK ON SFR ASTRID INNOVATIVE CORE: NEUTRONIC AND SAFETY TRANSIENTS SIMULATION

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Abstract

ASTRID is a fast reactor being designed by the CEA to achieve a level of safety that exceeds that of conventional fast reactors. In particular, an axially heterogeneous core with an upper sodium plenum is employed to achieve a non-positive sodium void reactivity worth. In order to address the simulation challenges for this innovative concept, the US Department of Energy's (DOE) Laboratories (Argonne National Laboratory and Idaho National Laboratory) and the CEA are performing neutronic and transient benchmark calculations for an ASTRID model based on design specifications provided by the CEA. The blind comparison of the initial DOE and CEA results are found to be in good agreement, enhancing confidence in CEA predictions of key ASTRID safety relevant parameters and transient behaviour. For several parameters, compared uncertainties in computed values are significant and further studies are needed to reduce them.

1. INTRODUCTION

The Advanced Sodium Technological Reactor for Industrial Demonstration (ASTRID) is a fast neutron reactor being designed by the Alternative Energies and Atomic Energy Commission (CEA) and its industrial partners to achieve a high level of safety [1]. In particular, an axially heterogeneous core with an upper sodium plenum is employed to achieve a near zero global sodium void worth and promote inherently safe behaviour of the core during unprotected transients. Under a framework agreement between United States of America and France for cooperation in low carbon energy technologies, the CEA and the US Department of Energy (DOE) are performing neutronics and transient benchmark calculations for a model of the current preliminary ASTRID design. The DOE contributions to the neutronic and transient calculations are being performed by the technical experts at the Idaho and Argonne National Laboratories, respectively.

The primary objective of the collaboration is an assessment of key neutronics performance parameters and safety characteristics of the specified ASTRID configuration for a limited set of transients:

- Comparison of calculated k_{eff}, key reactivity feedback coefficients, and power distributions including the uncertainties and the effects of these uncertainties;
- Comparison of safety margins for two anticipated transients without scram.

After an intense team effort devoted to performing the benchmark calculations independently by both sides based on specifications provided by the CEA, the first joint technical meeting was held in Saclay in December 2012. The results, included in this paper, reflect the blind comparisons of the calculations presented at the December meeting. Both the neutronics and transient analyses results obtained by the US and French teams were found to be in reasonably good agreement. For the integral sodium void worth, the discrepancy is found to be lower than \$0.5 with the calculated core power distribution, neutronics flux shape and reactivity feedback coefficients being quite similar. The results of the selected transient benchmarks for the unprotected loss of heat sink and station blackout scenarios also compare favourably, confirming the applicability of both DOE and CEA codes for analysis of the key safety characteristics of this novel, axially heterogeneous ASTRID core design.

2. ASTRID CORE BENCHMARK MODEL AND TRANSIENT SPECIFICATIONS

ASTRID is a 1500 MW(th), sodium cooled, oxide fuelled, pool type fast reactor prototype [1]. A primary objective of the design is to obtain inherent safety with sufficiently large margins to sodium boiling and core melting during unprotected accidents, but also to achieve a negative internal void worth against even more severe hypothetical accidents. A simplified benchmark model of ASTRID has been created by the CEA for comparative neutronics and transient analyses to confirm the key safety characteristics of the core and the plant design.

Several aspects of the complete ASTRID design, including the dedicated decay heat removal systems, are disregarded for this benchmark model.

The axial core configuration is illustrated in Fig. 1. This axially heterogeneous core configuration was designed to maximize the sodium leakage to achieve a negative overall sodium void worth. Two special zones are used to increase neutron leakage at the top of the core. First, a large sodium plenum is positioned immediately above the top of the core to increase leakage probability. Second, placing a fertile blanket zone in the axial centre of the inner core increases the flux at the top of the core.

The ASTRID core layout is illustrated in Fig. 2. The inner core region has 177 fuel assemblies each with an active core height of 1.1 m in two fissile and two fertile regions. The outer fuel region has 114 fuel assemblies with an active core height of 1.2 m. To increase the flux at the edge of the core, the outer fuel assemblies are slightly taller and do not have an inner axial blanket.

The schematic of the ASTRID's primary sodium heat transport system is illustrated in Fig. 3. The primary sodium system has three pumps and four intermediate heat exchangers. About 95% of the total flow from the pumps enters the core at 673 K and discharges at the top of the core at 823 K. The core outlet volume is part of the hot pool but, because it contains a large number of structural components, it is treated as a separate volume. Most of the sodium in the core outlet volume flows directly into the hot pool, but 10% of the sodium flows up through the control plug volume and washes over the control rod drives.

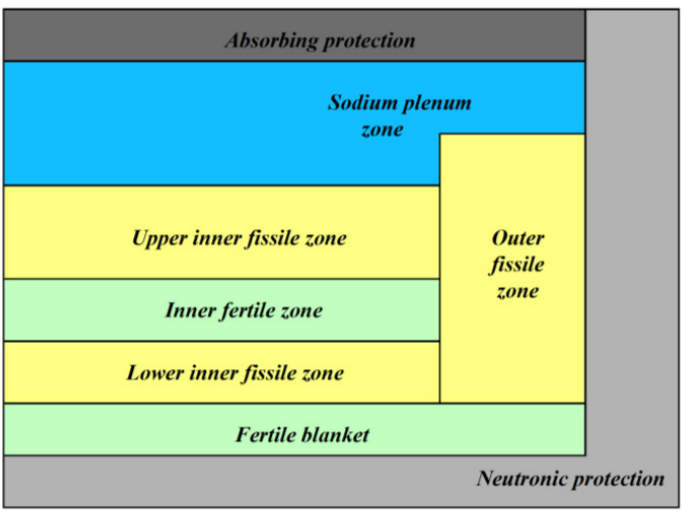

FIG. 1. Axial view of core.

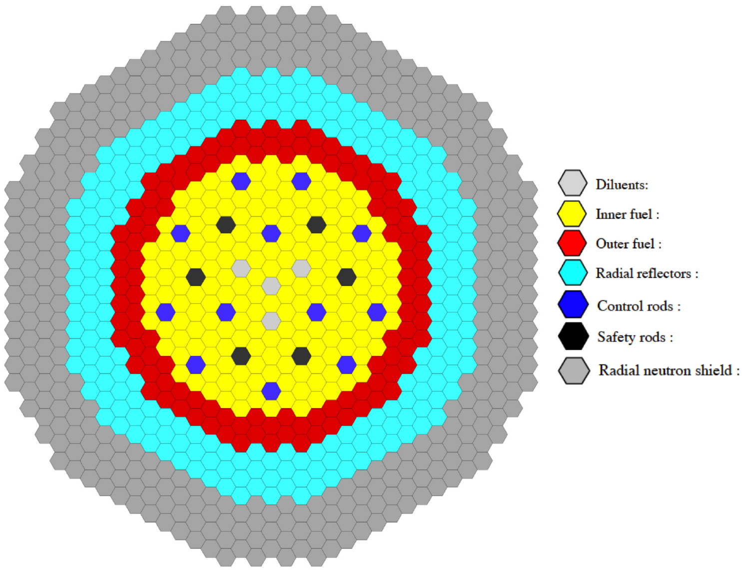

FIG. 2. Core layout.

In the diagrid, the 5% of total flow that does not enter the core flows upward along the reactor vessel wall through one of sixteen overflow pipes. This sodium then flows down and cools the outside of the hot pool shell, discharging back into the cold pool. For the benchmark model, the intermediate heat exchangers' intermediate side inlet temperature and sodium flow rate are specified as boundary conditions at 618 K and 1636 kg/s per intermediate heat exchanger, respectively. The rest of the intermediate sodium loop is ignored in the benchmarks.

Two unprotected transients have been defined for the comparative analyses. The first transient is an unprotected loss of heat sink (ULOHS), which is an unscrammed intermediate pump trip defined by the mass flow rate versus time values provided by the CEA. The second transient is an unprotected loss of supply station power (ULOSSP), which is an unscrammed station blackout with pump trips in both the primary and intermediate loops, also defined by pump speed versus time values provided by the CEA.

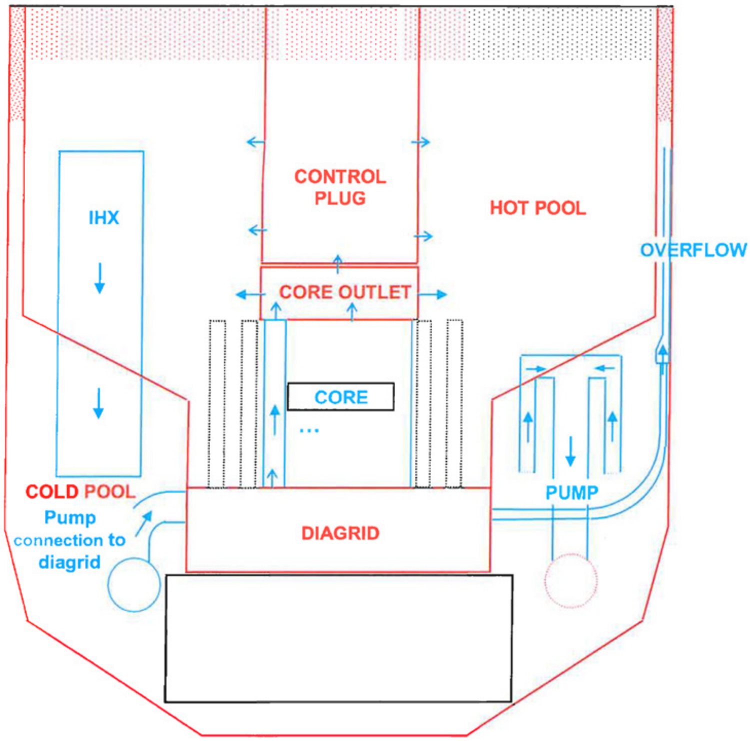

FIG. 3. Primary sodium system.

3. NEUTRONIC BENCHMARK

As part of the neutronics benchmark, the core multiplication factor, delayed neutron parameters, mass for main actinides, integral power per fuel assembly, maximal volumetric and linear power radial distributions, linear power axial profiles, maximal flux radial distribution, flux axial profiles, absorber rod bank worth and Doppler constants were evaluated independently by the teams at CEA and INL. The sodium void effect was evaluated for the following three fuel assembly conditions: (i) draining of the sodium plenum, upper gas plenum, upper pin plugs, fissile and axial inner blanket; (ii) draining of the sodium plenum, upper gas plenum, upper pin plugs and upper fissile and (iii) draining of the sodium plenum only. The reactivity feedback coefficients needed for the transients benchmarks were evaluated for the thermal expansion of sodium, fuel clad, wrapper, fuel/fertile pins and the diagrid.

The codes used by the CEA for the neutronics benchmarks include ERANOS [2], including SNATCH [3, 4] solver, and TRIPOLI [5], both using the JEFF3.1 nuclear data set. At INL, the neutron transport calculations have also been carried out with both stochastic (MCNP5 [6]) and deterministic codes (ECCO [7]/VARIANT [8]/BISTRO [9]/H3D-finite difference diffusion from the ERANOS code system). The INL team used the ENDF/B-VII.0 library as the reference set for neutron cross-sections, but some calculations have been repeated using the JEFF3.1 data and ENDF/B-VII.1. In both groups' calculations, the geometry has been represented with the maximum fidelity allowed (i.e. heterogeneous description of fuel assemblies) as shown in Fig. 4.

All calculations were performed using the ASTRID benchmark specifications for a BOL core configuration with all materials, geometries, and cross-section data at (i) reference parameters for 20ºC and (ii) for nominal operations conditions. In the latter case, the geometry was expanded and material densities adjusted to conserve mass. Most regions of the core are assumed at a temperature of 750 K, the lower axial blanket, with associated pins, clad, fuel assembly wrapper and sodium at 900 K, and the fuel and internal axial blanket, with associated pins, clad, S/A wrapper and sodium at 1500 K.

The US and French values for neutronics parameters are found to be in reasonably good agreement. A comparison of the global parameters at 20°C and at nominal power for a BOL core is provided in Table 1. The whole core sodium

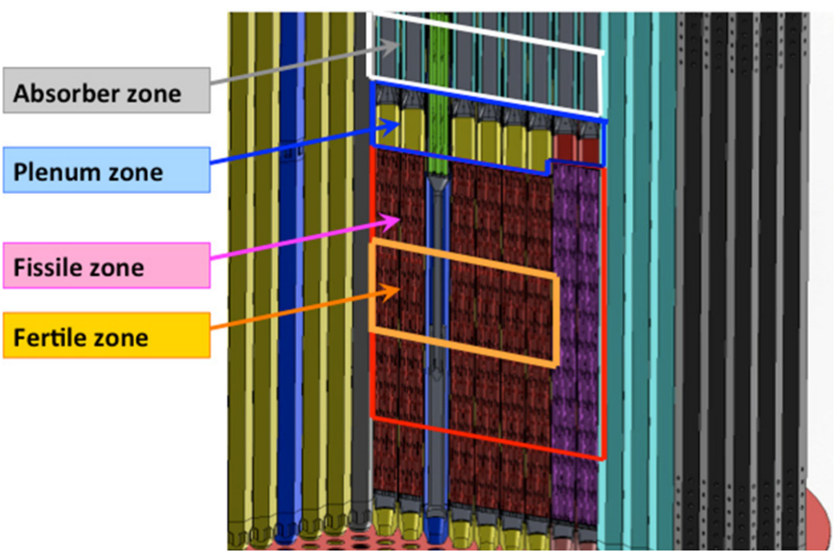

FIG. 4. ASTRID core layout considered in the neutronic benchmark.

void worth calculated by both teams are also compared in Table 2. A comparison of the zone dependent sodium void worth at nominal power for a BOL ASTRID core is provided in Table 3 for three separate voided configurations, as shown in Fig. 5: the whole core, the upper fissile zone + Na plenum, and Na plenum only.

TABLE 1. A COMPARISON OF CEA AND DOE (INL) RESULTS OF BOL CORE REACTIVITY (PCM)

	ERANOS – JEFF-3.1 CEA	MNCP – ENDF/BVII.0 DOE	DELTA CEA-DOE
20 °C	6764	6280	484
Nominal power	4640	4205	435
20°C → nominal power	2124	2075	49

TABLE 2. A COMPARISON OF CEA AND DOE (INL) RESULTS OF BOL WHOLE CORE SODIUM VOID WORTH

	ERANOS – JEFF-3.1 CEA	MNCP – ENDF/BVII.0 DOE	DELTA CEA-DOE
20 °C	–3.7$	–4.1$	0.4$
Nominal power	–2.9$	–3.3$	0.4$
20°C → nominal power	–0.8$	–0.8$	0

TABLE 3. A COMPARISON OF CEA AND DOE (INL) RESULTS OF BOL ZONE DEPENDENT SODIUM VOID WORTH AT NOMINAL POWER

Voided zone	ERANOS – JEFF-3.1 CEA	MNCP – ENDF/BVII.0 DOE	DELTA CEA-DOE
Whole core	–2.9$	–3.3$	0.4$
Upper fissile and plenum	–4.9$	–5.1$	0.2$
Plenum only	–5.1$	–5.4$	0.3$

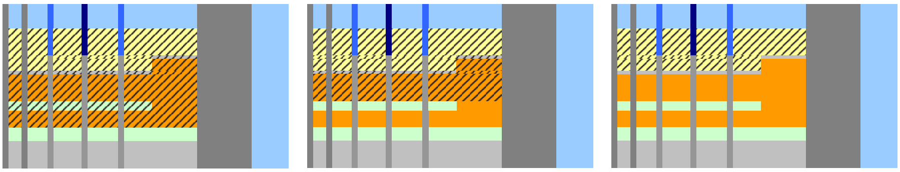

FIG. 5. Configurations for sodium void worth calculations (voided regions are shaded areas).

The results indicate that the library bias from cross-section data set differences (ENDF versus JEFF) as calculated with MCNP code is about 500 pcm and, for the sodium void worth, the maximum discrepancy is about \$0.4. Doppler (temperature) reactivity effects were also compared (see Table 4) and differences were found to be relatively small, assuring good consistency for the input data needed for the transient codes used in the subsequent safety analysis. The calculated power and neutronics flux shape and feedback coefficients are also quite similar, further indication of the consistency between US and French teams' results.

TABLE 4. A COMPARISON OF CEA AND DOE (INL) RESULTS OF BOL ZONE DOPPLER (TEMPERATURE) REACTIVITY EFFECTS AT NOMINAL POWER

	Doppler Effect (pcm)	
	ERANOS — JEFF-3.1 CEA	MNCP — ENDF/BVII.0 DOE
Everything +1000 K	–535	–494
Fuel at 2500 K	–338	–315
Blanket at 1900 K	–157	–149
Clad at 2500 and 1900 K	–19	–12
Wrapper at 2500 and 1900 K	–21	–18

In general, the observed differences stay within estimated uncertainties. A thorough, but still not exhaustive, uncertainty analysis was carried out by the two teams for the main integral parameters of interest, and nuclear data library effects were also studied in detail. In particular, the effects of use of JEFF3.1 (the reference library for the CEA) with respect to ENDF/V-VII.0 (the reference library for the DOE) were broken down in terms of isotopes, reactions and energy range using perturbation theory on a simplified R-Z model. A similar detailed analysis was performed by the two teams for the uncertainty evaluation where CEA used the COMAC [10] covariance matrix data and DOE used COMMARA 2.0 [11]. The main differences and uncertainties were found to be associated with ^{238}Pu (fission and capture), ^{239}Pu (fission), ^{23}Na (elastic and inelastic) and ^{238}U (inelastic). Table 5 summarizes data effects and uncertainties for k_{eff} and whole core sodium void worth.

TABLE 5. SUMMARY OF DATA BIAS AND UNCERTAINTY EVALUATION ON k_{eff} AND SODIUM VOID WORTH

Parameter (1σ)	Library bias	Uncertainties (DOE)	Uncertainties (CEA)
k_{eff}	500 pcm	1132 pcm	1434 pcm
$\Delta\rho_{Na}$	0.5$	0.5$	0.5$

The consistency of the results obtained with different nuclear data and codes is considered encouraging. However, the nuclear data uncertainties seem to dominate void worth estimates and need to be assessed, particularly for local behaviours. The total uncertainties and bias in void worth is estimated to be about $2 when considering a three sigma uncertainty range and conservative (not statistical) uncertainty component combinations. Although expected to be small, the consequences of these uncertainties on transient behaviours also need to be assessed.

4. TRANSIENT ANALYSES

The CEA and Argonne teams independently developed their transient analysis models for the ASTRID benchmarks using their CATHARE-2 [12] and SAS4A/SASSYS-1 [13] systems analysis codes, respectively. With both codes, the geometry of ASTRID primary heat transport system is represented by a series of perfectly mixed control volumes connected by liquid segments for modelling the sodium flow through the core, pumps, intermediate heat exchangers, control plug and overflow regions shown in Fig. 3. Although the ASTRID design includes dedicated decay heat removal systems in the primary coolant hot and cold pools, they are excluded in the benchmark specifications and the ultimate heat sink is assumed to be at constant ambient temperature outside the reactor vessel.

A single pin/channel model is assumed to characterize the fuel, coolant and structure of an average pin in a fuel assembly. The fuel assemblies with similar reactor physics and thermohydraulics characteristics are grouped together to form a channel. The initial SAS4A/SASSYS-1 model of the ASTRID core had two channels due to lack of details for the core flow distribution: Channel 1 represented the 177 inner fuel assemblies in an average sense, and Channel 2 represented the 114 outer fuel assemblies (i.e. the SAS4A/SASSYS-1 model currently calculates the peak temperature in two average fuel assemblies). The CATHARE-2 model of the ASTRID core has seven channels based on the estimated power density distributions in the core using the results of the neutronics benchmark. Because detailed design information for the other assembly types is not included in the benchmark, both the CATHARE-2 and SAS4A/SASSYS-1 core models are limited to the fuel assemblies.

The axial and radial power distributions, as well as the reactivity feedback coefficients for the transient benchmark models, come from the respective neutronics analyses in the France and the USA. Therefore, although estimated to be small, the transient analyses include the discrepancies in the neutronic benchmark. To obtain the flow distribution among the core channels, the mass flow in each channel is adjusted to give an outlet temperature of 823 K. Both the CATHARE-2 and SAS4A/SASSYS-1 reactivity feedback models include the Doppler, sodium density, fuel axial expansion and core radial expansion. The control rod driveline expansion reactivity feedback was also included in the CATHARE-2 model but excluded in the initial SAS4A/SASSYS-1 calculations due to lack of design information. With CATHARE-2, the radial core expansion is assessed solely on the basis of the diagrid expansion as a conservative approach, whereas with SAS4A/SASSYS-1, the expansion of the above-core load pads was also considered, leading to a less conservative assumption.

SAS4A/SASSYS-1 currently has the capability to model upper and lower blankets but not internal blankets. For these simulations, the inner axial blanket region is modelled as MOX fuel for thermohydraulic purposes. The net fuel expansion feedback effect in the inner blanket is expected to be small during the transients; therefore, neglecting it is not assumed to lead to a significant discrepancy.

The results from the independently created Argonne and CEA models were compared for the ULOHS and ULOSSP transient scenarios for the first time at a meeting held in December 2012, and these are presented here. The US and French results compare reasonably well with similar trends for power and temperature histories and consistent timing of peak fuel and coolant temperatures, confirming the applicability of both DOE and CEA codes for analysis of the ASTRID core. The estimated large margin to coolant boiling by both sides during both transients suggests that core damage or sodium voiding could be avoided for the ASTRID configuration studied in this benchmark.

During the ULOHS sequence, the early portion of the transient is driven by an increased inlet core temperature, which heats up the diagrid and causes a negative radial core expansion reactivity feedback. As power decreases, the negative radial expansion feedback is partially countered by a positive Doppler feedback as the core cools down. At around 100 seconds, the net reactivity reaches -6¢ for the Argonne model and -7¢ for the CEA model. Over the next 700 seconds, the net reactivity increases to -3¢, driven mostly by Doppler and also control rod driveline (CRDL) expansion in the CEA model. At around 800 seconds, net reactivity begins to decrease again as the diagrid continues to heat up, and at 3000 seconds it reaches -10¢ for the Argonne model and -9¢ for the CEA model.

As fission power decreases inherently during the ULOHS sequence, the core inlet and outlet temperatures converge, as shown in Fig. 6. Since the primary sodium coolant continues to remove the power at decay heat levels from the core at full flow, about 60 K difference between the asymptotic temperatures is attributed to differences in hydraulic resistances and rate of heat transfer to the ultimate heat sink (constant ambient temperature outside the reactor vessel). To better understand these differences, heat transfer between the hot and cold pools and the flow distribution between the core and overflow region are currently being evaluated and compared.

The CATHARE-2 and SAS4A/SASSYS-1 models also predict similar progressions for the ULOSSP transient. While the beginning of the ULOHS transient is driven by increasing core inlet temperatures, the ULOSSP transient is driven by an increase in power to flow ratio that approaches a peak of 3. As the flow rate levels off, elevated temperatures in the core keep the net reactivity negative, even as the Doppler reactivity feedback from the cooling fuel inserts ~80¢ of reactivity. This leads to a sudden drop in power level as shown in Fig. 7. While the SAS4A/SASSYS-1 model initially excluded the CRDL reactivity feedback, it predicts a more negative radial expansion reactivity feedback since it considers expansion of the above-core load pads. These compensation factors lead to a similar trend for the net reactivity between the CATHARE-2 and SAS4A/SASSYS-1 models as illustrated in Fig. 8 (the oscillatory appearance of the SAS4A/SASSYS-1 calculated net reactivity curve in Fig. 8 is due to the table look-up method for modelling pump coast-down).

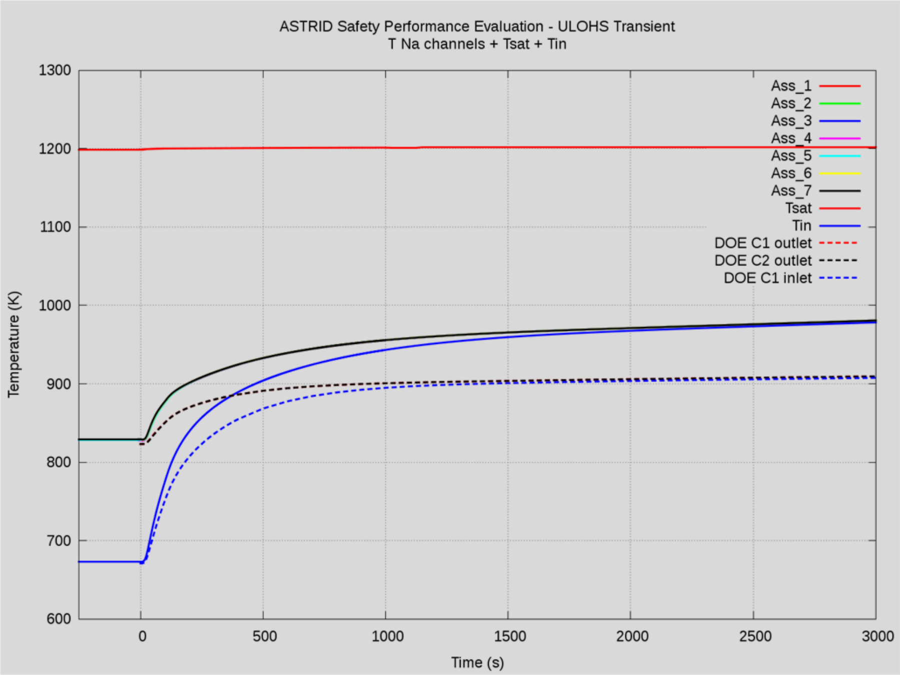

FIG. 6. ULOHS inlet and outlet sodium temperatures.

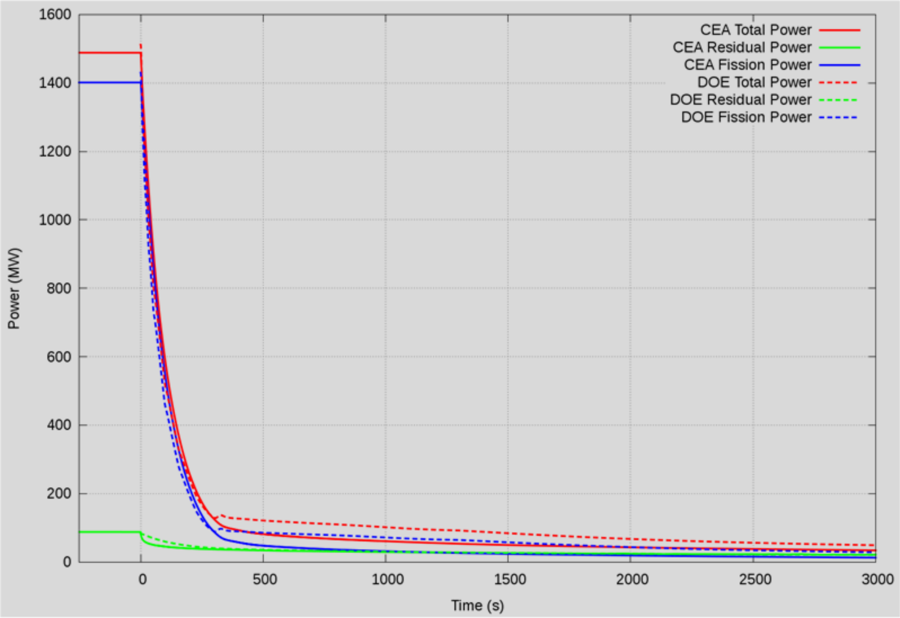

FIG. 7. Comparison of reactor power history for ULOSSP.

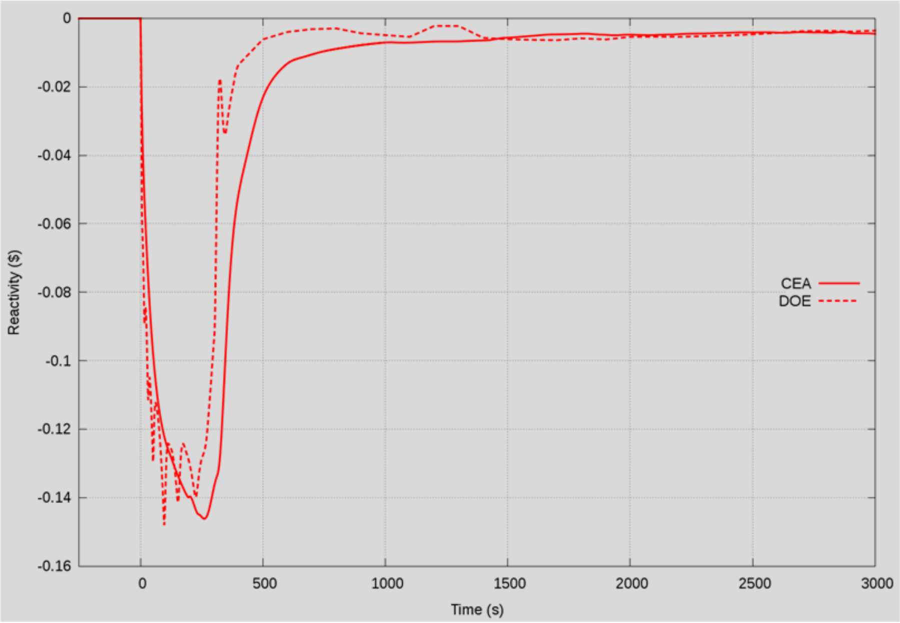

FIG. 8. Comparison of net reactivity history for ULOSSP.

Figure 9 illustrates the core inlet and outlet temperatures predicted by both models for the ULOSSP transient. Because the primary flow rate decays away, the inlet and outlet temperatures do not converge as in the ULOHS transient. As the cold pool heats up, both models predict the core inlet temperature would rise to ~800 K by 3000 seconds. Although the coolant outlet temperatures also follow a similar trend and reach peak temperatures at about the same time (around the 5 minute mark), the predicted margin to coolant boiling differs by about 50 K between the two models. The minimum margin to sodium boiling is predicted as 50 K with the CATHARE-2 model whereas the margin is greater than 100 K with the SAS4A/SASSYS-1 model.

Three modelling differences have been identified as the main drivers of the observed discrepancies in the US and French results:

- As the transient benchmark specifications do not provide the fuel thermal conductivity and heat capacity, the default oxide–fuel thermophysical properties in the CATHARE-2 and SAS4A/SASSYS-1 codes are assumed to be applicable. Potential differences in these default fuel properties will likely contribute to discrepancies in fuel temperature and Doppler reactivity feedback response during both transients analysed in this effort.

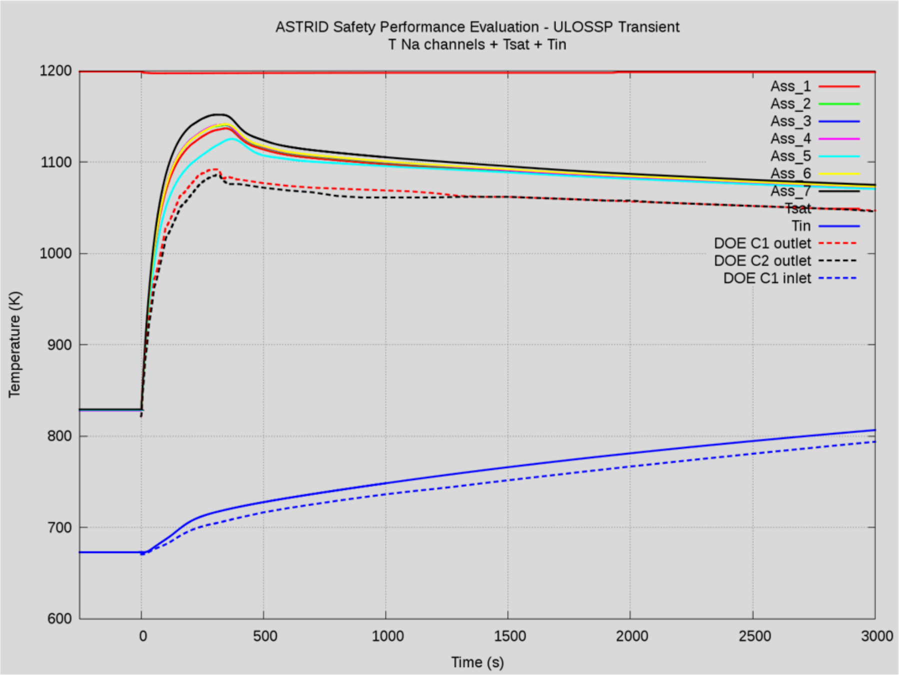

FIG. 9. ULOSSP inlet and outlet sodium temperatures.

- The CEA's CATHARE-2 model accounts for the differential expansion between the core, vessel, control rods and control rod drives (each driven by a different time constant) to include the CRDL expansion reactivity feedback. Owing to limited information in the benchmark specifications, the initial Argonne model with the SAS4A/SASSYS-1 code did not include this feedback effect. During the second revision of the Argonne calculations, a simple CRDL expansion model that only accounts for expansion of the control rod drives and the vessel walls is also included.
- With CATHARE-2, the radial core expansion is assessed solely based on the diagrid expansion whereas with SAS4A/SASSYS-1, the expansion of the above-core load pads was also considered leading to a less conservative assumption. During the second revision of the Argonne calculations, only the diagrid expansion is considered to be consistent with the CATHARE-2 model.

5. CONCLUSIONS

ASTRID is a sodium cooled fast reactor being designed by the CEA with a unique axially heterogeneous core and upper sodium plenum as the main features aimed at achieving a near zero (or negative) global sodium void worth. According to a framework agreement between the DOE and the CEA, the teams at CEA, INL and ANL performed neutronics and transient benchmark calculations to verify the basic characteristics of the ASTRID design.

The CEA and INL teams evaluated both deterministic and Monte Carlo methods for the neutronics benchmarks using ENDF/B as well as JEFF3.1 nuclear cross-section data for cross-comparisons. Generally, good agreement is observed between the CEA and DOE results for core multiplication factor, sodium void worth, axial linear power and flux distributions. However, based on the estimated uncertainties for separate components of void worth, the worst case combination of uncertainties could still lead to a slightly positive integral sodium void worth for the configuration studied in this benchmark. Therefore, additional modelling is recommended to reduce the uncertainties.

As with the comparison of blind calculations, the results of transient analyses are also found to be generally in good agreement, capturing the trends consistently and demonstrating the benign response of the ASTRID core for the both unprotected (without scram) transient sequences studied. The main difference in asymptotic coolant temperatures for the ULOHS case is attributed to differences in orifice coefficients, form losses and other flow resistances. The heat transfer from hot pool to cold pool, to the annular vessel cooling bypass flow region, and eventually to the constant vessel outer temperature is also identified as a potential source of uncertainty. In addition, about a 50 K difference in US and French results for the margin to boiling for the ULOSSP case is attributed to differences in implementation of radial core expansion feedback, CRDL expansion effect (ignored in the first round of DOE calculations due to lack of data) and fuel Doppler feedback due to potential difference in fuel properties and gap conductance model. These uncertainties are being evaluated during the second round of calculations.

The consistency between the independent DOE and CEA results for both the neutronics calculations and the transient analyses represents significant progress towards the stated objectives of this collaboration. Despite some differences in the assumptions made by both sides using the state of the art reference codes and methods based on best engineering judgement, the good agreement between the key core characteristics and transient behaviour enhances confidence in the CEA predictions of key ASTRID safety relevant parameters and behaviours. Proposed future activities are aimed at assessing (and reducing) the uncertainties arising from errors in computational models and data. A cost (or risk)/benefit analysis for very low sodium void worth cores is also considered important for both sides.

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MODELLING OF MULTI-PHYSICS PHENOMENA IN FAST REACTOR DESIGN: SAFETY AND EXPERIMENTAL VALIDATION

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Abstract

The paper provides a cursory look at current approaches in numerical modelling and simulation of typical multi-physics phenomena of concern relevant to the sodium cooled fast reactor design and safety. Emphasis is placed on the methods that are in practice and their verification and validation programmes, including for those of fluid–structure thermal interactions due to thermal striping, thermodynamics of sodium–water chemical reactions, multi-component and multi-phase flows in the fuel degradation and core meltdown phases. Several of the numerical simulations of these phenomena are shown with verification and validation programs that employ not only separate effect small scale experiments of clean geometry but also for large scale integral tests or mock-up experiments. The last part of this paper will be spent on discussions on a more quantitative validation basis with identification of errors and/or uncertainties based on the Bayesian rule.

1. INTRODUCTION

This paper takes a brief look at modelling practices for several multi-physics phenomena which are of concern and relevant to sodium cooled fast reactor design and safety. It includes those phenomena which involve single phase fluid–structure thermal interactions due to thermal striping, and multi-component multi-phase flow phenomena in sodium–water chemical reactions and those which involve numerous complicated and mixed transport mechanisms of different energetics. It also discusses the various timescales and microscopic to macroscopic scale phenomena, represented by the fuel damage and degradation as well as core meltdown and relocation under postulated severe core damage conditions. Focus is placed on typical numerical approaches to these phenomena that are in practical use and their verification and validation programs.

Most of the simulation of these multi-physics phenomena is carried out by large computer codes of complicated code structure. In principle, these codes are supposed to be well validated for separate effect small scale experiments of clean geometry, and therefore users of the computer code should be confident that constituents of the code are well verified and validated. To this degree, numerical simulation technology has become a major tool of safety evaluation. Nevertheless, in parallel, experiments are yet required to compensate where the simulation may not be able to delineate the phenomena or to enhance understanding of the physics.

All these multi-physics simulation models are being further subject to validation in comparison with so-called down scaled as well as large scale integral experiments on either out-of-pile or in-pile facilities. However, their validation for integral tests or mock-up experiments tends to be on the basis of qualitative judgement and is often criticized for its being subjective, not objective nor quantitative enough. The last part of this paper will be spent on possible approaches to surmount these criticisms with identification of errors and/or uncertainties inherent in computation based on the Bayesian rule.

2. FLUID–STRUCTURE THERMAL INTERACTIONS

2.1. Outline of high cycle thermal fatigue in the JSFR

High cycle thermal fatigue caused by thermal mixing phenomena has been one of the most important issues in the design and safety of the Japan sodium cooled fast reactor (JSFR) [1]. Figure 1 shows the target areas related to the thermal fatigue issues in the JSFR. In the reactor, the perforated plate called the core instruments plate is installed at the bottom of the upper internal structure in order to support thermocouples and the other sensors for operating and safety measures [2]. Below the core instruments plate, hot sodium comes from fuel subassemblies and cold sodium flows from both control rod channels and blanket fuel subassemblies located in the outer region of the core. There, the core instruments plate surface, the upper guide tubes and the control rod drive mechanisms are exposed to temperature fluctuations, resulting from fluids mixing, and thus experience possible thermal stress. Such a cyclic stress may cause crack initiation and crack growth in these components depending on the frequency characteristics and the amplitude of the temperature fluctuation.

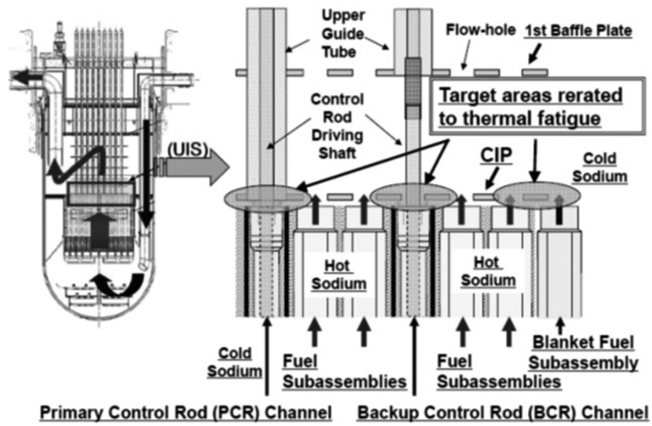


FIG. 1. Outline of thermal striping phenomena on the core instruments plate in the JSFR.

Numerical and experimental investigations on these thermal striping phenomena have been conducted at the Japan Atomic Energy Agency (JAEA).

2.2. Development of numerical estimation method and verification and validation procedure

A fluid–structure thermal interaction simulation code MUGTHES has been developed at the JAEA to investigate temperature fluctuation generation mechanisms and to estimate high cycle thermal fatigue in the structure of the JSFR [3]. The MUGTHES has two calculation modules for the unsteady thermohydraulic analysis and the structure unsteady heat conduction analysis to simulate the thermal interaction between flow and structure fields. The large eddy simulation approach is employed in the thermohydraulics module. In the verification and validation process [4], uncertainty quantification analysis was performed by using the grid convergence index method [5] referring to the guideline [6]. The verification, numerical schemes and discretization methods in the thermohydraulics analysis module are verified through the numerical simulations of fundamental problems at laminar flow conditions in the literature and also those in the structure analysis module and the conjugate heat transfer model are verified through the simulations of the theoretical unsteady heat conduction problems [7]. The validation has been conducted through the numerical simulations for fundamental turbulent flow problems in the literature and for typical element experiments for thermal mixing phenomena in the T-junction piping systems (T-pipes) and in the parallel triple jets tests.

2.3. Validation with thermal mixing problem in WATLON experiment

A typical example of validation of MUGTHES is shown with numerical results for thermal striping phenomena in a T-pipe junction. Table 1 gives boundary conditions at two typical flow patterns in the WATLON experiment [8]. The large eddy simulation with the standard Smagorinsky model was conducted [7]. Numerical results are validated in comparisons with the velocity and temperature measurement.

TABLE 1. BOUNDARY CONDITIONS OF WATLON EXPERIMENT

Case	Main pipe			Branch pipe		
	T_m (°C)	W_m (m/s)	Re (×10^5)	T_b (°C)	V_b (m/s)	Re (×10^5)
(a) Wall jet	48.0	1.46	3.8	33.0	1.0	0.66
(b) Impinging jet		0.23	0.5			

Figures 2(a) and (b) show the instantaneous fluid temperature distributions on the symmetric axial cross-section at the impinging jet and the wall jet cases, respectively. In the figures, the branch pipe jet at low temperature (T_b = 33°C) enters into the main pipe flow at high temperature (T_m = 48°C). Figures 3(a) and (b) show the instantaneous large scale vortex structures at the impinging jet and the wall jet conditions, respectively.

In the impinging jet case, as shown in Fig. 2(a), the magnitude of momentum inertia of the branch pipe jet was superior to that of the main pipe flow and then the branch pipe jet easily came into the main pipe. In front of the jet, wavy temperature boundaries appeared and the area of hot fluid lay in between the cold fluid areas. As shown in Fig. 3(a), large scale vortex structures existed in front of the branch pipe jet. The vortices, as a part of the branch pipe jet, conveyed cold fluid to the upper part of the main pipe and a striped temperature distribution, that is thermal striping, was caused on the upper surface.

In the wall jet case (see Fig. 2(b)), magnitude of momentum inertia of the main pipe flow was superior to that of the branch pipe jet. After the jet enters into the main pipe, a wavy temperature boundary appeared in the central part of the main pipe, above the wake. As shown in Fig. 3(b), large scale hairpin vortex structures appear behind the branch pipe jet. In comparison between Fig. 2(b) and Fig. 3(b), the tops of the hairpin vortex structure corresponded to those of the wavy temperature boundaries.

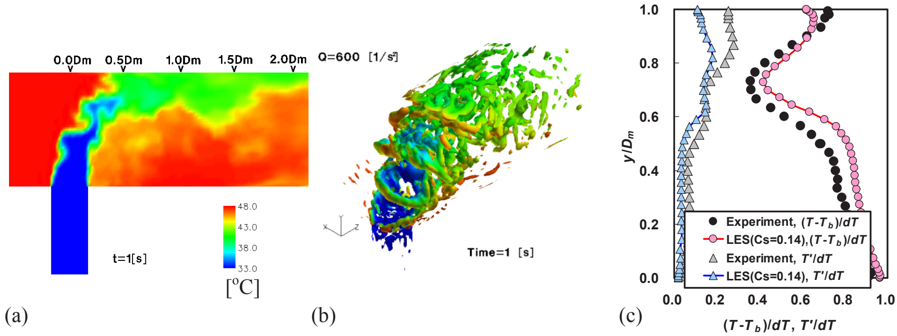


FIG. 2. Numerical results at impinging jet case. (a) Instantaneous distributions of fluid temperature on symmetric axial cross-section, (b) instantaneous distributions of large scale vortex structures and (c) time average and fluctuation intensity of fluid temperature in 0.5D_m downstream.

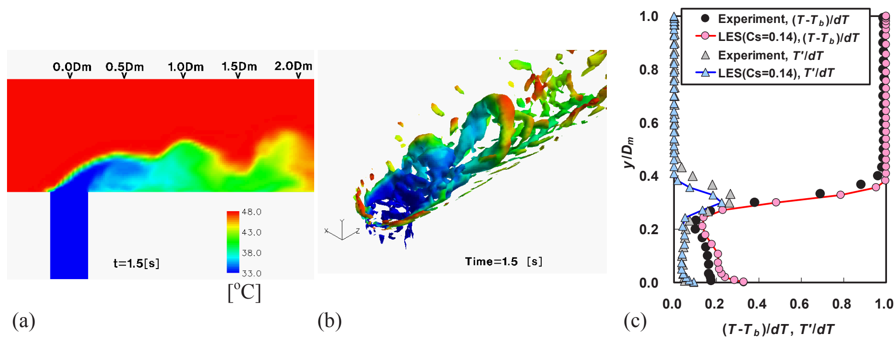


FIG. 3. Numerical results at wall jet case. (a) Instantaneous distributions of fluid temperature on symmetric axial cross-section, (b) instantaneous distributions of large scale vortex structures and (c) time average and fluctuation intensity of fluid temperature in 0.5D_m downstream.

In Fig. 2(c), vertical profiles of the time averaged fluid temperature and the temperature fluctuation intensity on the symmetric axial cross-section at 0.5D_m downstream from the connecting point at the impinging jet case were shown. The time averaged temperature difference (T-T_b) and the fluctuation intensity T' were normalized by the fluid temperature difference before mixing dT (= T_m–T_b). As for the time averaged temperature profile, the numerical result almost agreed with the experimental result. The temperature fluctuation intensity of the numerical result was almost half that of the experimental results in the upper part of the main pipe (y/D_m > 0.6). In Fig. 3(c), vertical profiles of the time averaged fluid temperature and the temperature fluctuation intensity in the wall jet case were shown. Not only the profiles of fluid temperature in the numerical simulation but also the fluctuation intensity were in good agreement with those of the experiment.

It is noted that the large scale vortices have a significant role in the thermal striping phenomena, i.e. the temperature fluctuation generation. Therefore, simulation of such large scale vortices is significant in evaluating thermal fatigues on the pipe surface.

3. MULTI-COMPONENT MULTI-PHASE FLOW PHENOMENA

3.1. Sodium–water reaction modelling

When the pressurized water or vapour leaks from a failed heat transfer tube of a steam generator (SG) of SFRs, a high velocity, high temperature and corrosive jet is formed, accompanied by sodium–water chemical reactions in the shell side (see Fig. 4). It is known that the reacting jet may cause wastage on adjacent tubes. This wastage is attributed to erosion, flow accelerated corrosion (FAC) or a combination of both. There is also a possibility that degradation of a mechanical strength from a temperature rise in the tube wall may cause an overheating rupture. Possible failure propagation may lead to damage expansion, replacement of the equipment and long term shutdown of the plant. Therefore, prevention of the failure propagation is a major concern in design of the SG.

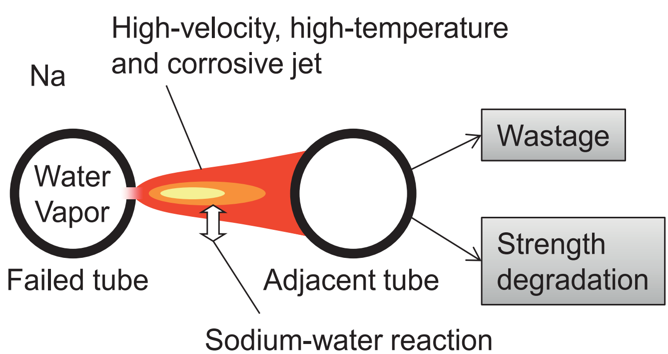


FIG. 4. Reacting jet under tube failure accident.

The computational approach has a great advantage over mock-up tests with respect to cost, development period and flexibility for changes of the design. To enable safety assessment of the SG, the SERAPHIM code for compressible multi-component multi-phase flows with sodium–water chemical reactions under tube failure accident conditions was developed [9]. The code predicts profiles of the velocity, temperature and concentration, which are necessary to evaluate the possibility of failure propagation. The code is based on a multi-fluid model considering compressibility to calculate the multi-phase flows consisting of water, liquid sodium and a multi-component gas. Transport of the species is calculated from the advection–diffusion equation. Models were newly developed for the chemical reactions at the interface between water vapour and liquid sodium, as shown in Fig. 5.

The surface reaction model calculates the mass generation rate by the above chemical reaction. An infinite reaction rate was assumed, implying that the progress of the chemical reaction at the gas–liquid interface is limited by the mass flow rate of the reactant gas towards the interface that is to be provided in

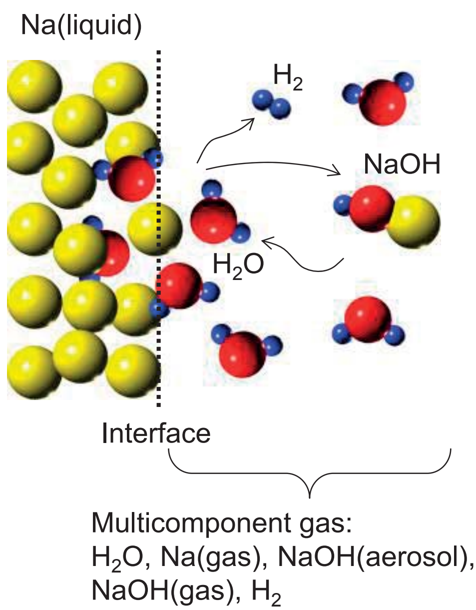


FIG. 5. Chemical reaction at the interface between water vapour and liquid sodium.

proportion to the interface heat transfer coefficient (10 000 W/m^2/K) [10], mass fraction of the water, interface area concentration, the Lewis number and inverse of the heat capacity and gas phase specific heat with empirical constant multiplier.

Applicability of the present numerical methods was confirmed through the analysis of various basic experiments such as the underexpanded gaseous jet in the liquid pool with chemical reaction [11]. Furthermore, the analysis of the SWAT-1R experiment [12] was performed to validate applicability to the actual situation in an SG. The integral test apparatus of the SWAT-1R experiment, originally designed to simulate the Monju SG configuration, is shown in Fig. 6 and its details are available in Ref. [12]. The water vapour of 17.0 MPa and 352°C was injected into the liquid sodium pool through the nozzle and reacted with the sodium. A forced flow of the liquid sodium was provided inside one tube (hatched dummy tube in Fig. 6) during the test in order to simulate the heat exchange with water in the actual system. The constant pressure condition was applied to both the inlet boundary and the upper end boundary which corresponds to the liquid sodium free surface.

The void fraction distributions at different timings after the beginning of leakage are shown in Fig. 7. Figure 8 shows a comparison of the temperature distribution on a vertical plane intersecting with the centre of the inlet boundary, where comparisons are made focusing on the high temperature region location, sizes, expansion and the peak temperature values. Discrepancies in the size of the high temperature region were observed but explained by the fact that in the

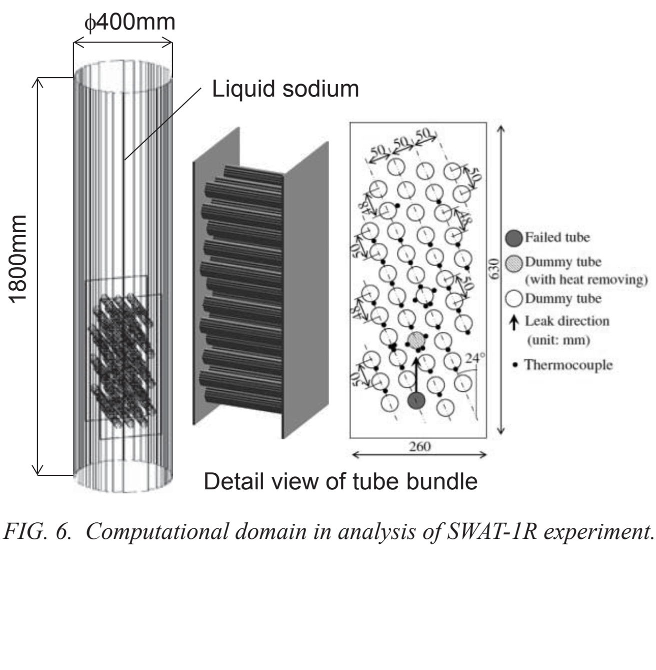


FIG. 6. Computational domain in analysis of SWAT-1R experiment.

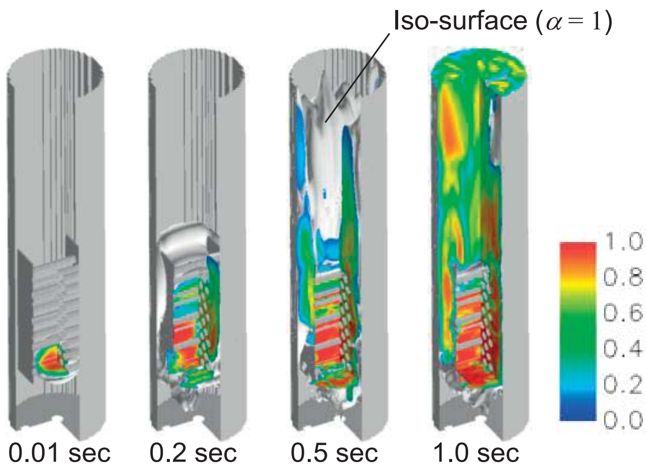


FIG. 7. Calculated void fraction.

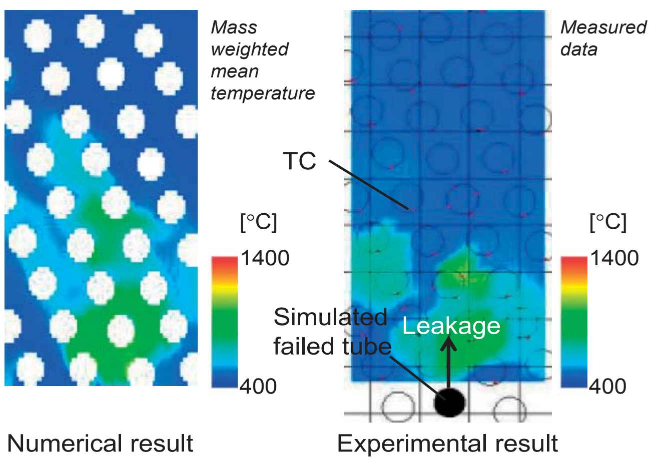


FIG. 8. Comparison of temperature distribution.

experiment, the reaction heat was removed by the liquid sodium flow inside the target tubes, whereas in the calculation, the tubes were treated as the adiabatic structure in the analysis.

3.2. Core disruptive accidents and fuel subassembly degradation modelling

Core disruptive accidents (CDA) of LMFRs or SFRs have been postulated in a category of the beyond design basis accidents in consideration of their extremely low occurrence frequency. However, in view of the severe accident issues triggered by the Fukushima Daiichi nuclear accident in March 2011, CDAs have attracted more attention than ever and should be re-evaluated in terms of the overall risks to consider in fast reactor safety. In this regard, multi-physics modelling of the extremely complicated core meltdown process should be revisited and the analytical capabilities of the SIMMER code [13] or similar tools should be extensively improved with further verification/validation programmes. For this purpose, it is strongly recommended that PIRT be carried out in the case of severe accidents at LMFRs in view of the current trend of design extension condition category, even if the CDAs may not be the design basis accidents.

Under severe accident conditions of LMFRs, the phase change and relocation of core materials are key phenomena that control the event progression of accident sequences. Reduced coolant flow causes sodium boiling, clad melting, relocation and freezing, and fuel melting. In general, because of the coolant temperature profile due to heat loss through the periphery of the subassembly wrapper tube wall, the initiation of local sodium boiling and clad melting is incoherent depending on the location in the subassembly. The incipient molten cladding moves along the still intact fuel pin surfaces, and subsequent fuel disruption forms a mixture of liquid and solid materials which also relocate in the remaining fuel pin subassembly. Under such accident conditions, the materials composing the fuel pin subassembly relocate in a three-dimensional space and freeze at a relatively cold part [14] .

Many analytical tools have been developed to analyse CDAs, including the SAS3D [15], SIMMER-II [13] and SIMMER-III [16] for a whole core analysis with neutronic coupling and QUSAR [17] and SURFASS [18] for subassembly accidents. These codes have been successfully used by many users to analyse each accident condition.

In general, SAS3D or SIMMER is not designed specifically for local fault nor subassembly accidents. Also, modelling of subassembly geometry is rather simple in these codes. For example, a fuel subassembly is treated in one dimension in SAS3D and two dimensions (r-z or x-y) in SIMMER-II. Further, two phase sodium boiling codes such as SABER [19] and SABENA [20] can analyse part of such accidents up to sodium boiling; they do not cover the phenomena beyond the dryout.

In recognition of these limits, a multi-component multi-phase analysis program, KAMUI, was developed [21]. The approach taken in the code is characterized by the subchannel analysis approach to represent a fuel pin array geometry and multi-component, multi-phase and multi-field representations of the fluids. Here, the multi-component refers to fuel, steel and sodium; multi-phase to solid, liquid and vapour and/or gas; and multi-field to a mixture of liquid sodium, liquid fuel and steel, solid particles, and vapour and/or gas. By this approach, the momentum and thermal interactions between the materials can be analysed and more detailed evaluation of in-pile experiments simulating subassembly accidents, such as SCARABEE [22], has been made possible. An advantage in using a code such as KAMUI is that the coolability of the faulted subassembly could be evaluated without excessive conservatism, owing to more realistic simulation of the strongly space dependent and time incoherent behaviours of the subassembly local accidents. As an example, one of the comparisons between calculation and experiment is shown in Figs 9 and 10, which illustrate a simulation result of the in-pile experiment SCARABEE BE+1, one of the 19 fuel pin subassembly total and instantaneous inlet blockage case of the important influences of non-uniform temperature distribution in a subassembly on the event sequences. It was clearly understood that the models of vapour–fluid interface friction, wall–fluid friction and interface heat transfer area correlation dominate the liquid film motion, timing of liquid sodium dryout, clad melting, etc. Figure 9 shows successful results in simulating the oscillatory behaviour of outlet flow. Figure 10 shows the axial and radial evolution of sodium boiling regions in the outermost and inner subchannels with and without heat loss consideration. In general, these figures show how good agreement is obtained, but in a subjective sense.

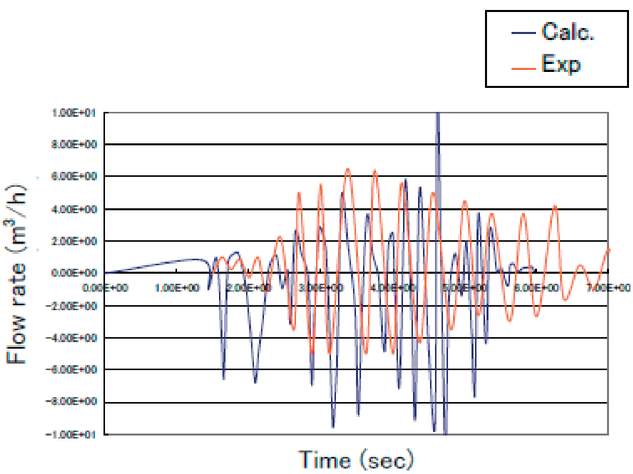


FIG. 9. Flow rate comparison at the subassembly outlet for total inlet blockage SCARABEE BE+1.

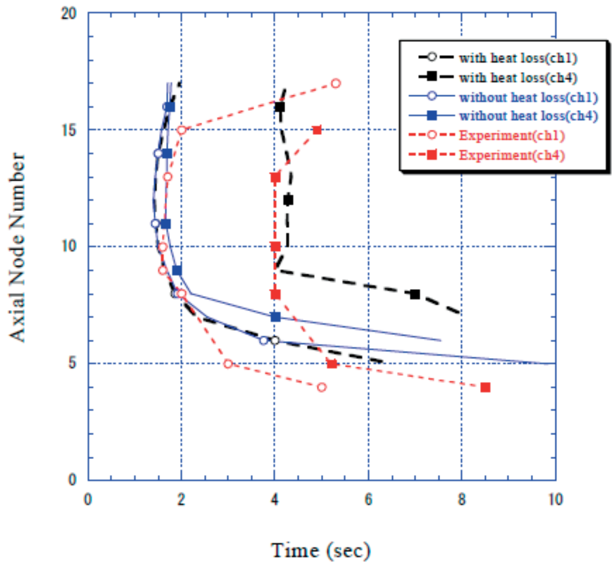


FIG. 10. Evolution of boiling region and the effect of heat loss (SCARABEE BE+1).

4. IDENTIFICATION AND ESTIMATION OF UNCERTAINTIES

Computer codes described in the preceding sections consist of many models representing the interactions between the variables through specific physical processes and producing results to be compared against experimental evidence. Generally, models are a result of approximations, sometimes compromises, excluding those processes which are considered negligible or assuming idealized conditions such as symmetry, periodicity or averaged behaviours in space and time. Certain packages, e.g. those of large eddy simulation or RSM turbulence models, are validated not only by subjective but by more objective methods, such as proper orthogonal decompositions [23]. Even if the models were perfect for representing the reality, however, their solution would not describe the reality unless a perfect knowledge of initial and boundary conditions were available, which is generally difficult to provide with high accuracy. Therefore, in analogy to experimental analysis, the result of a computer simulation represents one of the infinite solutions that could be realized, which would make it meaningless to provide its result alone without estimation of the uncertainty and error assessment.

In addition, agreement of computer simulation results has always relied on subjective interpretation of the results with experimental data. The method shown below represents one of the attempts to quantify the agreement with quantitative and visual comparisons that are not only visual image comparisons. Hereafter, a recently developed methodology [24] for model uncertainty quantification is briefly explained and its application is described in the field of sodium fast breeder reactors.

4.1. Uncertainty quantification method: Bayes Theorem

The method hereafter explained hinges on the concept of the Bayesian theorem, where the probability of a model to correctly represent the reality is updated by the continuous comparison with experimental evidence. The simplest form of Bayes rule is shown in the following equation:

$$P(B|A)=\frac{P(A|B)P(B)}{P(A)}$$

The equation simply states that the probability of an event B, i.e. P(B), can be updated given the information of a related event A taking place. In the case of nuclear numerical application, the Bayesian rule can be interpreted as: prior probability or the subjective degree of belief of a certain calculation model, P(B) can be updated to P(B|A) with P(A), and probability of A by experiment and P(A|B). Note that the likelihood P(A|B) is given a priori, but is subjective and, therefore, could be improved, for example, by the optimum estimation theory.

Carrying out as many calculations as possible with different model parameter values provides P(B|A) and any statistical information on the parameters: mean, standard deviation, and tolerance limits giving information on model precision, role of model components to be employed in risk analyses or decision making.

4.2. Example of application: Stratification in sodium flow

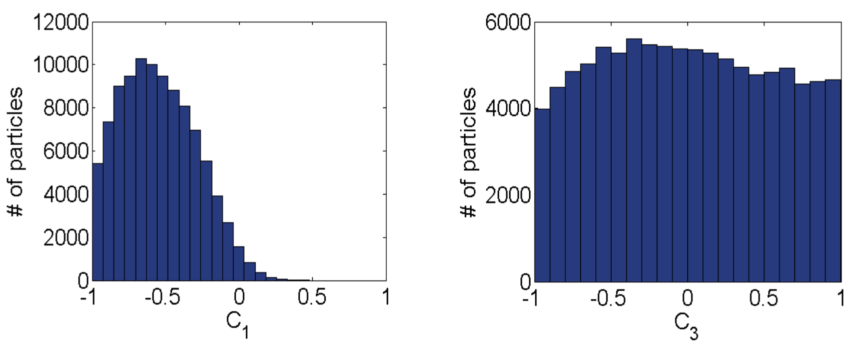
Stratified flows exist in sodium reactors after postulated pump trip and scram of the reactor. In the past, various studies were performed for the upper plenum, where buoyancy forces are predominant compared to the advection, so that stratification is introduced. Owing to the particular geometry of the plenum and flows of various temperatures from the blanket and driver fuels, stratification is likely to be transferred inside the pipe and, to some extent, affect the flow and pressure drop in a certain portion of the piping system.

Stratification is a well known physical phenomenon leading to erroneous prediction of velocity and temperature profiles in the mixing layer with turbulence models that neglect the gravity effect. Here, the effect of a more complete turbulent model, namely the algebraic heat flux model (AHFM) [25], is referred to as a reference. The AHFM consists of a new closure of the turbulent heat flux model which is expressed through two contributions, velocity gradient and gravity, in other words C_1 and C_3. The experiment [26] provides eight locations for temperature and velocity profile with an error of 10% due to the instrumentation employed. In addition, turbulent heat flux is provided not from direct measurements but from the quantity definition so that the error was estimated as 25%.

The methodology consists of running several simulations (depending on the number of the parameters and problem linearity [24]) adopting different values of the model coefficients to build a response surface. Thereafter, through the employment of a Monte Carlo method, also known as a particle filter [27], it is possible to find out a distribution around the optimized value, standard deviation and tolerance limits. These values are shown graphically in Fig. 11.

Figure 11 shows the probability density distribution of the two model components which resemble a Gaussian curve in a normalized frame. The mean value represents the optimized value of the coefficient taken into consideration, while the standard deviation can be interpreted as the subjective degree of belief of the turbulence model contribution (in this case, the velocity gradient and gravity components).

From the above results, it is visually clear that for the particular problem considered, stratification in sodium flow, the degree of belief of the first contribution shows a smaller standard deviation representing a strong degree to believe that the mean value of C_1 is suitable for the experimental prediction. In comparison, weaker confidence exists on the C_3 parameter.



a) C_1 velocity gradient b) C_3 gravity

FIG. 11. Probability density distribution of AHFM components.

4.3. Discussions

Beyond giving optimized values for each parameter and standard deviation, the combination of standard deviations extrapolates the degree of belief of the whole model. It is well known that probability is hard to be interpreted in an absolute frame, so that the briefly explained methodology perfectly fits in the frame of model or code comparison, for the choice of the model/code that represents the prediction of quantities of interests best.

In general, validation of a model or code is given in relation to how much the analyst is satisfied from the direct but subjective comparison of calculation with experiment. It is clear that an absolutely definite answer to the correctness of the code is not possible and eventually a subjective judgement cannot be ruled out, but this judgement should be made so as not to overshadow the process. The present discussion provides a method to postpone the introduction of subjective decision until quantification is developed so that:

- Further subjective decisions could be made more rationally;
- Relative comparisons between physical models with more or fewer uncertainties involved would become more meaningful and justified on a quantitative basis.

5. CONCLUSIONS

In this paper, we have focused on the current practices of numerical modelling and simulations of thermohydraulic phenomena in sodium cooled fast reactor systems and verification/validation programs for modelling single phase fluid–structure thermal interactions due to thermal striping and multi-component, multi-phase thermofluid dynamics of sodium–water chemical reactions and core meltdown process, as well as fuel degradation modelling in subchannel analysis. All these multi-physics simulation models are subject to validation in comparison, as a prerequisite step, with separate effect small scale experiments of clean geometry. A subsequent step requires so-called down scaled integral experiments on either out-of-pile or in-pile facilities.

It has been pointed out that, in practice, validation by large scale integral tests or mock-up experiments as shown in this paper for engineering multi-physics phenomena is likely to be made on rather a qualitative basis, often relying on many subjective judgements. In validation processes, although an eventual subjective judgement cannot be ruled out, these should be minimized. To make it more quantitative and rational, a proposal has been made, in reference to thermal stratification phenomena of sodium flows, for the identification of errors and/or uncertainties inherent in computations based on the Bayesian rule.

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EXPERIMENTAL STUDIES FOR A SAFETY GRADE DECAY REMOVAL SYSTEM FOR THE PROTOTYPE FAST BREEDER REACTOR

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Abstract

In the present scenario, the decay heat removal from nuclear reactors is of great importance. In the sodium cooled pool type fast breeder reactors, decay heat generated from the reactor core followed by an unprotected reactor trip is transferred to the sodium pool initially by forced circulation and then by the buoyancy driven flow. The heat is removed from the sodium pool by means of dedicated passive decay heat removal systems. To demonstrate and validate the different decay heat removal mechanisms in the Prototype Fast Breeder Reactor (PFBR), different studies have been conducted in sodium and in water medium. The heat removal mechanism from the core to pool and then to the decay heat exchanger was studied in a 1:4 scale water model of the reactor. The effect of interwrapper flow on decay heat removal was studied separately with a full scale slab model of the reactor core in water. Further, the decay heat removal from the sodium hot pool to the atmosphere through the sodium to sodium decay heat exchanger and sodium to air heat exchanger was studied with a scaled down sodium model of the system in a facility called SADHANA. The methodology of the experimental study on decay heat removal for the PFBR and the results obtained are discussed in this paper.

1. INTRODUCTION

Successful operation and deployment of the fast breeder reactor forms the second and important part of India’s three stage nuclear programme. Presently, the Prototype Fast Breeder Reactor (PFBR), of 500 MW(e), is at an advanced stage of construction at Kalpakkam. Decay heat removal is very important for the safe operation of the fast breeder reactors. Considering the present global scenario, it is essential to ensure safe and reliable decay heat removal. In the PFBR, in addition to decay heat removal by the normal heat transport path, one more passive decay heat removal arrangement has been provided to achieve high reliability in decay heat removal. This passive decay heat removal system is a safety grade decay heat removal (SGDHR) system which removes the decay heat generated inside the core by natural circulation. This makes it an important safety system in the case of station blackout or when the normal heat transport path is unavailable [1]. Schematic representation of the SGDHR along with the normal heat transport path is shown in Fig. 1. The SGDHR system in the PFBR consists of 4 independent loops, each 8 of MW(th) heat removal capacity. Each loop consists of a sodium to sodium shell and tube heat exchangers (DHX), secondary sodium loop, sodium to air heat exchanger, chimney and damper. The SGDHR system works on the principle of heat removal by natural circulation and hence does not require any active system except dampers provided on the air side. The decay heat generated inside the core is removed by natural convection flow that is developed due to core–hot pool–DHX interaction. The heat transported to the secondary side of the DHX from the hot pool is removed through sodium to the air heat exchanger, which is connected to a natural draught chimney, and finally heat is rejected into the atmosphere, which is the ultimate heat sink.

Natural convection induced flow is quite susceptible to the various conditions prevailing in the reactor and hence demonstration of the onset of natural circulation is important to ensure higher reliability of the entire SGDHR system [2]. Experimental as well as analytical studies were conducted to understand the core thermohydraulics under natural circulation conditions. The SGDHR is a complicated natural circulation system with complex flow fields. The CFD analysis of the complete system with simulation of the influential local effects is an extremely difficult task. Ideally, it is better to conduct studies on a complete SGDHR system. However, simulation of the whole SGDHR system in

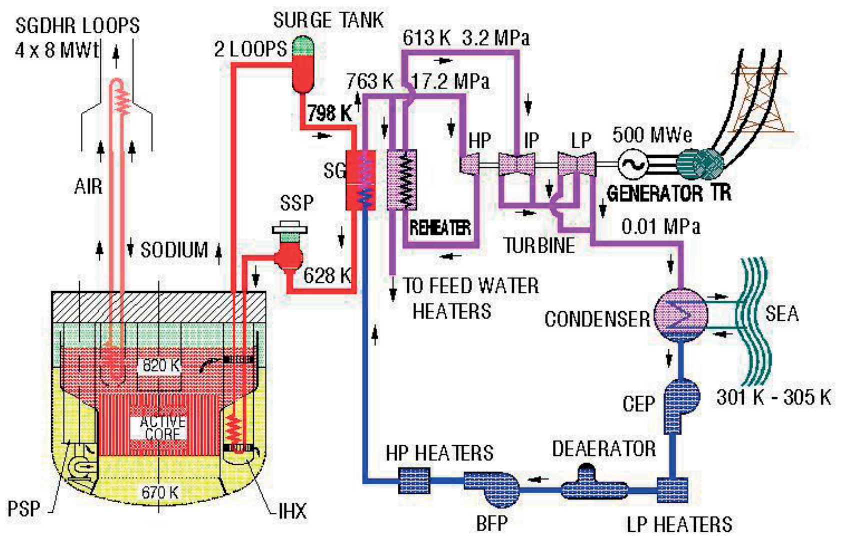


FIG. 1. Heat transport paths available in the PFBR.

a single model and with sodium as a simulant is complex and not economically viable. Hence, total system performance has been studied as a combination of experiments carried out using water and sodium. The complete decay heat removal system, comprising DHX–pool interaction, intermediate loop and air side heat removal have been studied using different experimental models. The DHX–core–hot pool interaction during the SGDHR condition has been carried out in the SAMRAT model (1:4 scale model of the PFBR primary circuit). These studies have demonstrated core cooling under natural circulation and DHX–core–hot pool interaction during this process. Probable natural circulation paths during SGDHR operation are shown in Fig. 2. The experimental studies were carried out under different configurations to analyse the effectiveness of these natural circulation paths available in the reactor. During these studies, it was felt necessary to study the core heat removal by interwrapper flow (IWF) in detail. IWF is caused when cold sodium exiting from the DHX mixes with the hot pool and penetrates into the gap regions between the subassemblies (SAs) (interwrapper spaces) and cavities present in the outer region of the core and enhances natural circulation [3]. Detailed experiments were carried out using a 1:1 scale slab model of the reactor core and hot pool to demonstrate the existence of IWF.

The establishment and behaviour of a natural circulation path in the secondary loop was studied in greater detail in a sodium loop (SADHANA loop). The role of the water and sodium experimental studies in development of the decay heat removal system was primarily to demonstrate core cooling under natural convection and to generate a database for code validation. Performance

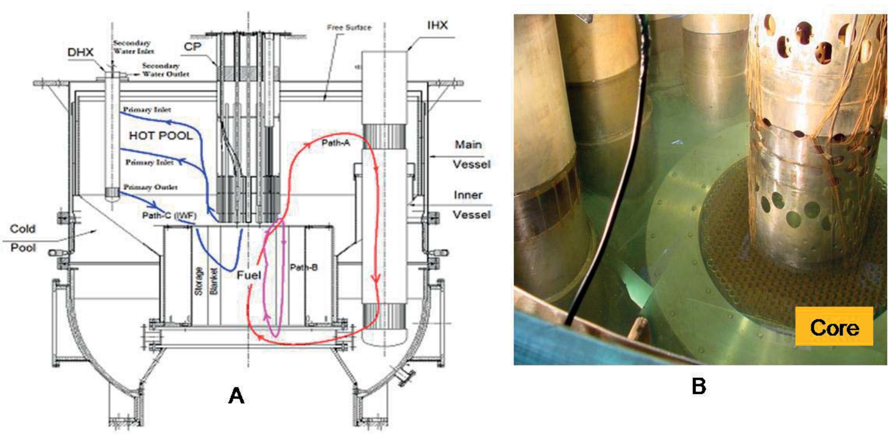


FIG. 2. (a) Natural convection flow paths during decay heat removal in the pool, (b) SAMRAT model internal components with core SA and interwrapper spaces.

of the complete SGDHR system was investigated and natural convection heat removal was demonstrated using all the above tests. This paper elucidates all the experimental studies carried out for PFBR SGDHR system.

2. SIMILARITY CRITERIA

Geometric and dynamic similitude is essential to maintain the prototype condition in the model. Geometric similitude can be maintained in the scaled down model of the reactor. Dynamic similitude is required to maintain a balance between the forces acting on the prototype. This can be achieved by maintaining the governing non-dimensional numbers between model and prototype. Non-dimensional numbers obtained from the normalized governing differential equations for SGDHR operation are given below [4].

Assuming that the working fluid is incompressible and the Boussinesq approximation is valid, the governing one dimensional equation for the conservations of mass, momentum and energy can be written as follows:

$$\frac{\partial W}{\partial s} = 0, \tag{1}$$

$$\frac{L}{A}\frac{\mathrm{d}W}{\mathrm{d}t} - g\oint \rho(z)\mathrm{d}z + \left(\frac{fL}{D} + K\right)\frac{W^2}{2A^2\rho_r} = 0, \tag{2}$$

$$\frac{\partial T}{\partial t} + \frac{W}{\rho_r A}\left(\frac{\partial T}{\partial s}\right) = \begin{cases} \dfrac{UA_s(T-T_w)}{A_c\rho_r C_p}, & \text{for convective heater} \\ -\dfrac{UA_s(T-T_w)}{A_c\rho_r C_p}, & \text{for pipe} \\ -\dfrac{UA_s(T-T_w)}{A_c\rho_r C_p}, & \text{for convective cooler} \end{cases} \tag{3}$$

When the system is at steady state, that is $\frac{\mathrm{d}W}{\mathrm{d}t} = 0$, the momentum conservation equation reduces to

$$\frac{g}{\rho_r}\oint \rho(z)\mathrm{d}z-\left(\frac{fL}{D}+K\right)\frac{v^2}{2}=0 \tag{4}$$

At steady state, by neglecting the heat loss from the pipe, the cold leg and hot leg temperatures will be constant and the difference of this is ΔT. Then

$$\frac{g}{\rho_r}\oint \rho(z)\mathrm{d}z \text{ will reduce to } \mathrm{g}\,\beta\,\Delta TH \text{ and hence,}$$
$$\mathrm{g}\,\beta\,\Delta TH-\left(\frac{fL}{D}+K\right)\frac{v^2}{2}=0 \tag{5}$$

Equation (4) states that the buoyancy head developed in the system due to the temperature difference is completely utilized by the frictional resistance and viscous dissipation to develop a mass flow rate of W at steady state conditions. The characteristics of the system are determined by the ratio of the forces in the first and second terms of Eq. (4). From Eq. (5) it can be understood that the non-dimensional group which is characterizing the behaviour of the natural circulation loop at steady state is the Richardson number (Ri) and Euler's number (Eu). Other important non-dimensional numbers to be considered while simulating the steady state phenomenon are the Peclet number (Pe) and Reynolds number (Re). The Peclet number characterizes the heat transfer and the Reynolds number characterizes the heat and momentum transport in the system. All the similarity criteria can be perfectly achieved only in a full scaled model with sodium as the working fluid. However, sodium experimental studies are difficult to carry out and incur huge costs. Therefore, experimental studies associated with DHX–core–hot pool have been carried out in a scaled down model of the reactor and water has been used as the simulant. Intermediate circuit and AHX heat removal studies are carried out with sodium as simulant. It is observed that the most important number to be simulated for the natural circulation driven SGDHR study is the Ri number [5]. Hence, studies have been carried out using Ri similitude with distortion in Pe and Re [6]. Table 1 lists the Pe distortion and Re number distortion with Ri number simulation for water and sodium studies. Even though the results obtained from the water studies cannot be transposed to reactor conditions directly, results will be of great use in validating the analytical code for the actual conditions.

TABLE 1. SIMILARITY ASPECTS OF DIFFERENT SCALED MODEL

Scale	(Ri^*)	(Re^*)	(Pe^*)	(Eu^*)
Water (SAMRAT model, 1:4 scale)	1	0.023	20.4	1
Sodium (SADHANA facility)[a]	1	0.168	0.168	1

$Re^* = \frac{Re_m}{Re_p}$, $Ri^* = \frac{Ri_m}{Ri_p}$ and $Pe^* = \frac{Pe_m}{Pe_p}$ $Eu^* \frac{(Eu)_m}{(Eu)_p}$

Subscript ‘m’ and ‘p’ denotes model and prototype respectively.

[a] Since the SADHANA facility is not a geometrically similar model as the SAMRAT, the reference points for evaluating Re is secondary piping and for Pe it is the heat exchanger tubes.

3. STUDIES CARRIED OUT WITH WATER

Core–pool–DHX interaction performance studies have been carried out using water in the SAMRAT model. The interwrapper flow is an important parameter to ascertain core coolability under the SGDHR condition. These studies have also been carried out using water in a slab model of the PFBR primary circuit. Details of these water tests are discussed in the following paragraphs.

3.1. SAMRAT model decay heat removal studies

SAMRAT (ScAled Model for ReActor Thermohydraulics) is a 1:4 scaled model of the PFBR primary circuit. A schematic view of the SAMRAT model is shown in Fig. 2. This model simulates all major components in the primary circuits of the PFBR which are essential to study the thermohydraulic behaviour in the hot pool and the cold pool, i.e. main vessel, inner vessel, thermal baffles, intermediate heat exchangers (IHX), DHX, pumps, control plug, core assembly, etc. The model core is divided into three main parts: fuel zone, blanket zone and storage zone. The outer shielding and reflector region subassemblies (SAs) of the core are simulated by annular shells. Each individual SA, simulated in fuel, blanket and storage regions, are in the form of circular sleeves fixed in the grid box top and bottom plate by a threaded joint. Design of blanket and storage SA’s are also similar to the fuel SA. Decay heat removal by natural circulation has been studied in this model, particularly experiments associated with DHX–core–hot pool–cold pool interaction during SGDHR operation. Probable dominant natural circulation paths envisaged in the reactor during SGDHR operation are also shown in Fig. 2. Path A indicates natural circulation through hot pool–IHX–cold pool–core–hot pool. Path B indicates natural circulation through hot pool–outer SAs(core)–grid box–fuel SA (core). Path C is the flow path through interwrapper spaces, i.e. interwrapper flow. The dominant natural circulation paths available in the prototype are simulated in this model.

3.1.1. Experimental methodology

Heaters were provided in the model fuel SAs to simulate the decay heat generation inside the core. Heater power was calculated on the basis of the Ri simulation and pressure drop in a SA is simulated by Eu simulation. Heater rods are provided with a heated length which simulates the heating region of the PFBR core. Four DHX are immersed in the hot pool of the SAMRAT model for removal of simulated decay heat. All four DHX are multi-row straight tube counter-current type heat exchangers and dipped into the hot pool of the model at respective locations. Decay heat removal in the DHX is simulated through forced circulation on the secondary side of the DHX. Experimental studies were carried out under different experimental conditions. Important locations for the temperature measurements were the hot pool, core and interwrapper spaces. Thermocouple racks with thermocouples positioned on them were used for temperature measurement in the hot pool and in interwrapper spaces. In the interwrapper spaces, thermocouples were provided in radial direction and axial direction to the core. Thermocouples were also provided at other important locations such as the DHX primary inlet–outlet, DHX secondary inlet–outlet, core inlet, IHX inlet window, etc. A rota-meter was used for the flow measurement at the secondary side of the DHX.

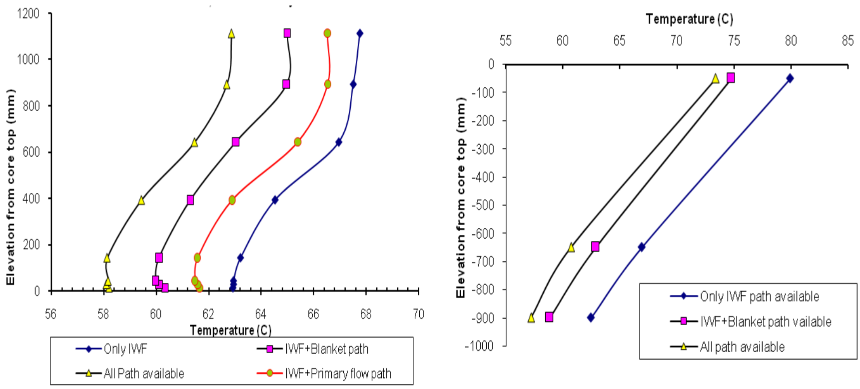
Experiments were carried out mainly to demonstrate the onset of natural convection and to see the effect of various natural circulation paths on the overall core heat removal. Experiments were started with and without DHX operation in the loop to understand the effect on core cooling. Studies were dedicated towards understanding the influence of the various heat removal paths available in the reactor on the core cooling. Initially, flow through IHX was blocked, which enabled core coolability through IWF and reverse flow through blanket SA. Studies were also conducted to understand the heat removal by the IWF path alone, where inflow through the IHX and inter-SA flow were blocked. These comparative studies were very important in understanding the heat removal by different flow paths.

3.1.2. Experimental results

Experimental studies conducted with and without DHX in operation reveal its significant effect on core cooling. It is found that in studying the effect of DHX operation on central SA (CSA) outlet temperature [7], steady state was attained in a few hours after DHX started, and when experiments were conducted without DHX operation, temperatures appeared to be rising continuously. This study demonstrates the onset of natural circulation and concludes that DHX operation has considerable impact on the hot pool temperature pattern. Experimental studies were carried out to understand the effect of various decay heat removal paths in the reactor. The steady state hot pool temperature between the free surface to core top is shown in Fig. 3(a) [7]. It is evident from this figure that temperature difference across the hot pool for different experimental conditions appears to be almost constant. However, steady state hot pool temperatures are different under different conditions. Comparison of the steady state temperature of the core outlet and the hot pool temperatures shows that steady state temperature is lowest in the case where all heat removal paths are available and highest in the case of only the IWF path being availabile for heat removal. Heat removal by reverse flow through blanket and storage is also equally effective compared with other heat removal flow paths. Temperature pattern in interwrapper spaces was one of the important measurement aspects of these studies. The temperature profile in the interwrapper space across the core elevation for the heated zone (i.e. in fuel zone) is shown in Fig. 3(b) [7]. It can be concluded from the temperature profiles that there is no radial diffusion of heat from the fuel zone (heated zone) to the non-heated zone. The temperature gradient in the fuel zone with heat removal by only the IWF path is slightly higher than the temperature gradient indicated in other cases [7]. Hence, heat removal by IWF is comparable to the normal heat transport path. Hence, contribution by IWF on core cooling cannot be discarded under adverse conditions.

3.2. IWF studies in slab model of reactor

Experimental studies were conducted to understand flow through the interwrapper spaces in detail. A 1:1 scale slab model of the reactor core and hot pool was selected for this study. The large size of the model was useful in achieving the detailed geometric modelling in the core region. This also helped in instrumentation of the SA interwrapper space regions.



(a) Hot pool temperature profile (b) Temperature profile in IWS-1 (typ.)

FIG. 3. Temperature profiles during SGDHR condition.

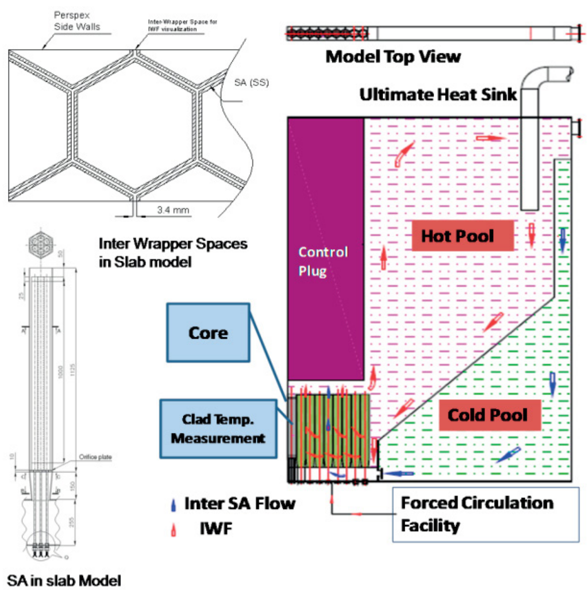


FIG. 4. Model schematic for IWF visualization study.

3.2.1. Experimental methodology

All geometrical features that are important in influencing the IWF have been considered in this model. The schematic of the model is shown in Fig. 4.

Only fuel and blanket zone SAs were simulated in the 1:1 scale of the prototype SA. Pressure drop was simulated with Eu criteria. The height of the core in the model is 1 m, which is the same as the active height of the core in the prototype. DHX heat removal is modelled by direct injection of cold water in the hot pool. The height of the liquid column above the core has been maintained. The side walls of the models are made up of transparent acrylic sheet at required locations to facilitate flow visualization. Rod type heaters in bundle form are used for simulating decay heat in the model. Heater power was estimated based on Ri similitude. Thermocouples, along with the data acquisition system, were employed for temperature measurement inside the model at various locations. Flow visualization studies were conducted by dye injection and also by 2-D particle image velocimetry recording. The experimental set-up was heated with model core heaters simulating the decay heat generated inside the core. DHX heat removal is simulated by cold water injection. This additional water influx leads to the overflow of similar amounts of water from the overflow line. This experimental procedure was continued until a steady state temperature was achieved in the model. Thermocouples located at different locations in the model monitored the temperature data at set time intervals and the IWF path was traced by a flow visualization exercise. Flow visualization and particle image velocimetry measurement were carried out separately and only after the system attained a steady state condition.

3.2.2. Flow visualization

Initially, dye was injected in the bottom region of the interwrapper spaces, above the top grid plate. Dye was seen to be coming out of all interwrapper spaces associated with heated SAs, demonstrating the presence of IWF during heat removal. Radial flow was also witnessed in the bottom conical portion of the SAs. This radial flow in the lower region is due to lower resistance to the flow path resulting from the conical shape of the SAs. This conical portion of the SA helps to penetrate the cold liquid emerging from the DHX exit towards the central core region and enhances the IWF. In the second phase of the flow visualization study, injection of the dye inside the SA through respective ports showed the presence of flow through the SA. In this case, dye was also observed to be emerging from all the heated SAs. It was possible to visualize a plume of hot water emerging from the SA and interwrapper spaces. These results prove the presence of IWF owing to the onset of natural convection.

3.2.3. Experimental results

Experimental test runs were carried out under different conditions and temperature measurements were carried out at important locations and particle image velocimetry measurements were also recorded during some of the studies.

Experimental studies were conducted for different cold water injection rates and at different overflow heights of the model. Cold water injection at the DHX outlet location indicates the different level of stratification in the hot pool. The steady state hot pool temperature is found to be lower with a higher cold water injection rate, i.e. with higher stratification in the hot pool. In addition to this, particle image velocimetry measurements were also recorded at the outlet of the fuel SA. The velocity profile obtained at the outlet of the SA is used for approximate estimation of the flow rate through the SA, which, in turn, is used for estimation of heat removal by flow through the SA. The approximate contribution of the IWF in core heat removal is 25%, as per particle image velocimetry measurement.

4. SODIUM EXPERIMENTS FOR THE SGDHR SYSTEM

The SADHANA(SAfety grade Decay HeAt removal in NAtrium) loop was set up to demonstrate and evaluate the natural circulation flow and heat removal in all three different heat removal paths, i.e. by primary pool, by DHX–AHX sodium circuit and by natural draught through a chimney for air circulation in the AHX (Fig. 5). This scaled down model of the circuit was designed, fabricated, installed and commissioned in IGCAR. The 1:22 scale model facility is based on Ri similitude. The capacity of the SADHANA loop is 355 kW and the height difference between the thermal centres of the DHX and AHX is 19.5 m.

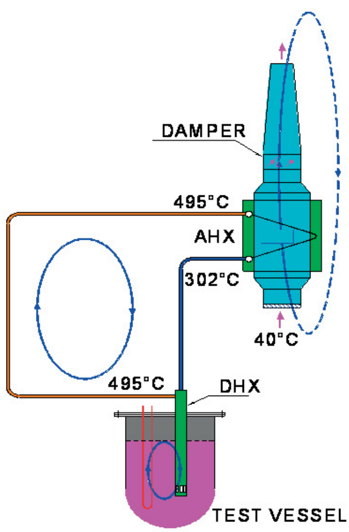


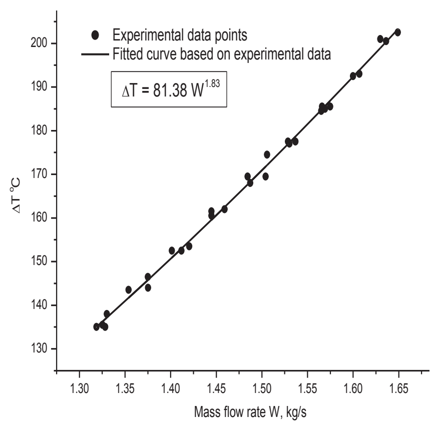
FIG. 5. Schematic of natural circulation path.

In SADHANA, the sodium in-test vessel IV, which simulates the hot pool of the PFBR, is heated by immersion type electrical heaters. This heat is transferred to the secondary sodium through the model DHX. The heat from the secondary sodium circuit is rejected to the atmosphere through the AHX. A 20 m high chimney develops the draught required to transfer the heat from secondary sodium to the atmosphere through the AHX.

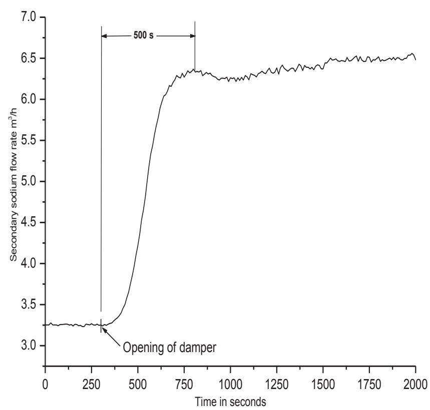
Different sodium pool temperatures can be achieved by controlling the input heater power through immersion heaters in the sodium pool. The design temperature of the in-test vessel IV, secondary sodium loop and related equipment is 600°C.

4.1. Experiments and results

Experiments in the SADHANA were conducted at different sodium pool temperatures. At a sodium pool temperature of 550°C, the secondary loops removes 19.4% more power than its rated capacity. Figure 6(a) shows the natural circulation as a function of the difference in hot and cold leg temperatures. As per one dimensional design calculations of the SADHANA loop at a pool temperature of 526°C, the secondary loop was expected to remove 355 kW with a sodium flow of 6 m^3/h in the secondary. At this condition, the experimental observations show a heat removal of 396 kW with a sodium flow of 6.55 m^3/h.



(a) Steady state Natural Circulation flow in SADHANA

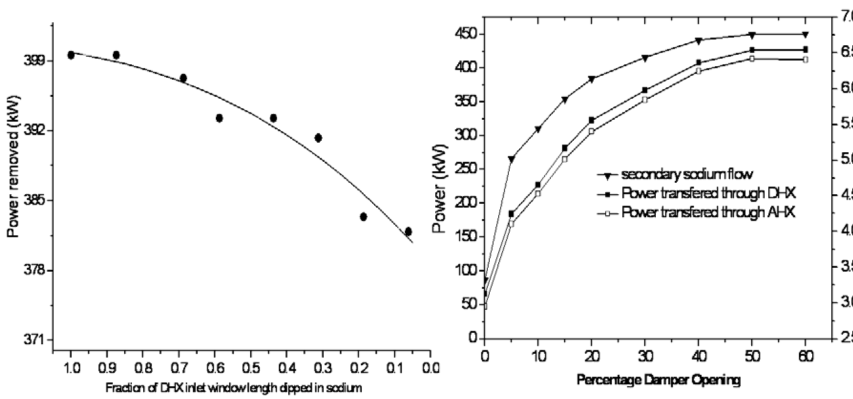


(b) Sodium flow when the AHX dampers opened with pool temperature of 550 °C

FIG. 6. Steady state experimental results.

The temperature drop/gain in the AHX and DHX remained close to the predictions. This study establishes the adequacy of different margins provided in the process design of the components and the system. The response of the SGDHR secondary sodium system on sudden opening of the AHX outlet damper was studied and it was found that the system would be fully functional around 510 seconds after initiation of the opening of the dampers. The setting up of sodium flow, air flow and various temperatures after the sudden opening of the damper were smooth and have not indicated major oscillations. The evolution of sodium flow followed by the damper opening is shown in Fig. 6(b). The heat transport of the SGDHR system under low sodium levels in the hot pool was also studied in the SADHANA facility. The low sodium level in the pool will result in a reduction of the effective heat transfer area of the DHX in the perforated inlet window region. The results from these studies have indicated that a 88% reduction in sodium level at the inlet window region causes a 2% reduction in secondary flow and a 5% reduction in power transported by the system. The reduction in the heat transfer area occurs in the region where the heat transfer from the primary sodium to secondary sodium is less effective. The variation in power transported by the system with respect to the primary sodium levels is shown in Fig. 7(a).

The heat transport capability of the SADHANA facility was evaluated under different partial damper opening conditions. At lower damper openings, such as less than 30% damper opening, the increase in power transport with respect to the increase in damper opening is high. At 50% damper opening, the heat transport capability of the system is stabilized and no further appreciable change has been observed, as shown in Fig. 7(b).



(a) Pool level drop studies **(b) Heat removal capability at diff. damper opening**

FIG. 7. Steady state experimental results.

5. CONCLUSION

The viability of the fully passive PFBR decay heat removal system was successfully demonstrated by experiments with a water medium in different facilities and with sodium in the SADHANA facility. These experiments have great significance towards establishing fast reactor safety. Experimental water studies were carried out to understand the core–hot pool–DHX interaction during SGDHR operation in scale models of the PFBR. In a 1:4 scale model of the reactor, experiments were carried out to demonstrate decay heat removal and to understand the influence of complicated phenomenon such as IWF. Experimental results have shown that all the proposed heat transport paths have influence on core cooling. Comparative studies of different configurations have revealed that IWF is an effective heat transport path and reverse flow though blanket and storage SA also contributes towards core cooling. A flow visualization exercise in a separate slab model of the reactor successfully demonstrated the presence of IWF during SGDHR operation. An approximate estimation of the IWF contribution is found to be 25% of the total heat removal. Performance of the SGDHR system during steady state, transients and some of the off-normal conditions were studied and characterized by the in-sodium experiments conducted in the SADHANA facility. These experiments revealed the adequacy and capability of the SGDHR system to remove the decay heat from the fast breeder reactor core after its shutdown.

NOMENCLATURE

A	Area
Cp	Specific heat at constant pressure
D	diameter of pipe
Eu	Euler Number
f	Friction factor
g	Acceleration due to gravity
H	Height
K	Pressure loss coefficient
k	Thermal conductivity
L	Length
Pe	Peclet number
Pr	Prandtl number
Re	Reynolds number
Ri	Richardson Number
s	spatial coordinate
T	temperature
t	Time
U	Overall heat transfer coefficient
v	Velocity of the working fluid
W	mass flow rate
z	coordinate along the height
β	Volumetric expansion coefficient
ΔT	Temperature difference
μ	Dynamic viscosity of the fluid
ρ	density

Subscripts:

^{c} Cross-sectional

^{s} Surface per unit length

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DEVELOPING A COMPREHENSIVE SOFTWARE SUITE FOR ADVANCED REACTOR PERFORMANCE AND SAFETY ANALYSIS

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Abstract

This paper provides an introduction to the reactor analysis capabilities of the nuclear power reactor simulation tools that are being developed as part of the US Department of Energy's Nuclear Energy Advanced Modeling and Simulation (NEAMS) Toolkit. The NEAMS Toolkit is an integrated suite of multiphysics simulation tools that leverage high performance computing to reduce uncertainty in the prediction of the performance and safety of advanced reactor and fuel designs. The toolkit effort is composed of two major components, the fuels product line, which provides tools for fuel performance analysis, and the reactor product line, which provides tools for reactor performance and safety analysis. This paper presents an overview of the NEAMS reactor product line development effort.

1. INTRODUCTION

The Nuclear Energy Advanced Modeling and Simulation (NEAMS) Program [1] of the US Department of Energy's (DOE's) Office of Nuclear Energy (NE) is developing a suite of advanced reactor and fuel simulation capabilities. The NEAMS Toolkit leverages current high performance computing capacity in the United States of America to provide predictions of reactor and fuel performance with unprecedented fidelity. While the Consortium for Advanced Simulations of Light Water Reactors project in the USA [2] and the Nuclear Reactor Integrated

Simulation Project in the European Union [3] are focused on the deployment of advanced simulation tools for the current fleet of light water reactors (LWRs), the NEAMS Program is focused on the development of advanced simulation tools to support the design and development of the advanced reactor types identified by the Generation IV programme. The NEAMS Toolkit consists of two primary components, the fuels product line and the reactor product line (RPL)

The objective of the NEAMS RPL is to enable the design of future nuclear power stations and reactor cores that implement enhanced safety and security features, produce power more cost effectively, and use natural resources more efficiently. To accomplish this goal, a significant shift in the approach to optimization of new core and plant designs is needed. The NEAMS RPL seeks to provide a suite of simulation tools that:

- Reduces margins resulting from predictive uncertainty by using mechanistic models and high fidelity simulation methods to increase accuracy and bridge gaps in experimental data and operating experience;
- Enables designers to reduce design margins by providing tools that eliminate the need for geometric simplifications and material homogenization in simulations and limits dependence on engineering correlations that have a small range of applicability; and
- Introduces opportunities for a new level of global optimization of the reactor/fuel system, especially for new reactor or fuel concepts, through integrated (concurrent or hierarchical) predictions of reactor and fuel performance.

To accomplish these goals, the RPL must enable users to integrate simulations of physical phenomena on three levels. Multiphysics integration provides connectivity between different physics modules. Multiscale integration provides connectivity between different scales of simulation within a single physics area (e.g. integration of multidimensional computational fluid dynamics (CFD) simulations of a single power plant component with one-dimensional lumped parameter simulations of the remainder of the plant). Multiresolution integration (or hierarchical coupling) allows information from a predictive, mechanistic simulation to be used to inform models of lower resolution and fidelity (e.g. using the results of a direct numerical simulation CFD analysis in place of a traditional correlation in a system level lumped parameter model).

2. REQUIREMENTS

NEAMS Program requirements are defined through interactions with the leading and principal investigators of the DOE-NE's reactor and fuel cycle technology research and development (R&D) programmes, as well as potential academic and industrial users of the NEAMS RPL. Consideration is given to three major requirement categories: (i) programmatic requirements, which result from the desire to align the NEAMS Program with the efforts of the DOE-NE R&D programmes, (ii) functional requirements, which are imposed by the anticipated workflow of the NEAMS RPL's end users and (iii) applications requirements, which are imposed by the target use cases to which the NEAMS RPL is expected to be applied. Of course, consideration is also given to funding requirements and limitations in developing the NEAMS Toolkit project plan.

2.1. Programmatic requirements

The NEAMS project works to support all of the DOE-NE's reactor and fuel cycle R&D programmes. As a result, the scope of the NEAMS RPL focuses on development of reactor-technology-neutral capabilities and includes support for a range of advanced reactor types:

- Sodium cooled fast reactors (SFRs);
- Prismatic gas cooled reactors (PMRs);
- Pebble bed gas cooled reactors;
- High temperature fluoride salt cooled reactors;
- Lead cooled fast reactors;
- Advanced LWRs.

NEAMS is focused on mid-term to long term deployment options, primarily the development of capabilities that leverage advanced simulation methodologies and existing high performance computing infrastructure. The programme is currently working towards a 2018 release date for the initial user-ready toolkit.

2.2. Functional requirements

Key functional requirements have been identified through end user meetings to focus and prioritize the scope of the development effort. These requirements characterize a spectrum of user expectations for the NEAMS RPL.

There is no single characteristic user for NEAMS RPL, and the toolkit must be able to support users who:

- Provide design and safety evaluations of current reactors for industry or R&D on advanced reactor designs for the DOE;
- Seek to complete both high fidelity single physics simulations and integrated multiphysics simulations of an entire plant;
- Have broad experience/expertise in all of the physics included in a multiphysics reactor simulation or have more limited exposure in only one area; or
- Use conventional multicore processor desktop workstations, commodity Linux clusters, or leadership scale petaflop computing facilities.

As a consequence, the NEAMS RPL must be able to meet a wide range of functional requirements. In response to this challenge, a modern, modular architecture has been adopted to enable the flexibility needed for development of a reactor-technology-neutral toolkit that can be customized to fit an individual end user's desired model complexity. However, some common functional requirements extend to all or large groups of potential users, including user interface and user support requirements. Functional requirements for any software suite must be expected to evolve as the sophistication of high performance computing hardware — and its users — evolves. For this reason, regular stakeholders' meetings are held to assess the consistency of development priorities with end user expectations [4].

2.3. Application requirements

Application requirements, which include the list of physical phenomena that must be represented in the toolkit, are derived from the key use cases that the software suite is designed to support. Target use cases should require accurate representation of the interaction of multiple physics models, be difficult to model correctly using conventional correlation based approaches, and support the development of multiple reactor technologies targeted by the programmes of the DOE-NE. Based on stakeholders' input, four target use cases have been identified to drive the initial development efforts:

(i) Evaluation of passive safety features resulting from multiphysics, multiscale reactor dynamics during unprotected loss of flow transients in SFR cores;
(ii) Identification of thermal striping and stratification in outlet plena and other large volumes, especially in SFRs and PMRs;

(iii) Assessment of natural convection stability during startup of advanced reactor designs with limited pumping capacity, especially advanced small modular reactors;
(iv) Prediction of core bypass flow impacts on core performance and safety, especially in very high temperature reactor cores using graphite pebble or PMR designs.

While no single use case covers the full range of reactor types included in the NEAMS RPL Toolkit's scope, the use cases identified do share some common features. All of the use cases are inherently multiphysics and require contributions from some combination of neutronics, thermal, fluid and structural mechanics models. They are also transient in nature, requiring that the physics modules resolve temporal changes in the fields that they simulate. Additionally, each of the use cases is tied to a complex geometry that has a strong influence on the relevant physical phenomena.

While consideration is given to all of the use cases in the development of the toolkit, initial demonstration and validation efforts focus on analysis of the passive safety features of the SFR in an unprotected loss of flow transient. In this transient analysis, the accurate prediction of feedback resulting from thermal expansion and mechanical distortion of the core structure is essential to accurate prediction of core power [5]. Therefore, the analysis requires a model that is inherently multiphysics and accounts for the complex geometry of the core and surrounding structure. Conventional methods accomplish this task using models that are carefully calibrated to available operating experience and separate effects data [6]. The NEAMS Toolkit seeks to provide an assessment capability that can be more easily extended to new reactor concepts. In particular, the toolset should be applicable to evaluation of the passive safety characteristics of new SFR designs resulting from multiphysics thermal-structural-neutronics phenomena, such as those demonstrated in the shutdown heat removal tests at the Experimental Breeder Reactor II (EBR-II) [7].

The SFR passive safety transient focus was selected as a first challenge for a number of reasons. Current state of the art codes cannot easily be used to address this problem for innovative reactor designs because the range of applicability of the empirical correlations used is limited. The problem is inherently multiphysics and requires integration of thermal, fluid, structural and neutronics phenomena. This is particularly true because the evolution of these different phenomena occur over similar timescales, requiring the codes to be coupled in a single simulation to accurately capture their respective dependencies. The problem is also inherently multidimensional and dependent on detailed representations of core geometry. The capabilities required are common to many reactor analyses, and the tools needed are extensible to other reactor types. Perhaps most importantly, the DOE owns significant, relevant experimental data for validation as a result of the development and operation of the EBR-II and the Fast Flux Test Facility.

3. THE NEAMS RPL SOFTWARE DESIGN

The NEAMS RPL is comprised of two major products that can be applied independently or in concert — the lumped parameter RELAP-7 reactor system simulation code and the high fidelity SHARP reactor core simulation suite [8]. To provide the flexibility needed to address the wide range of users and applications, highly modular software architecture has been adopted. The basic components and associated connectivity functions that make up the NEAMS RPL, illustrated in Fig. 1, are described below.

3.1. Neutronics modules

The NEAMS neutronics tools, built using the PROTEUS package as a foundation, provide a complete analysis capability including three dimensional transport, cross-section processing, reactor kinetics and depletion. They also include four major modules, as described in the next sections.

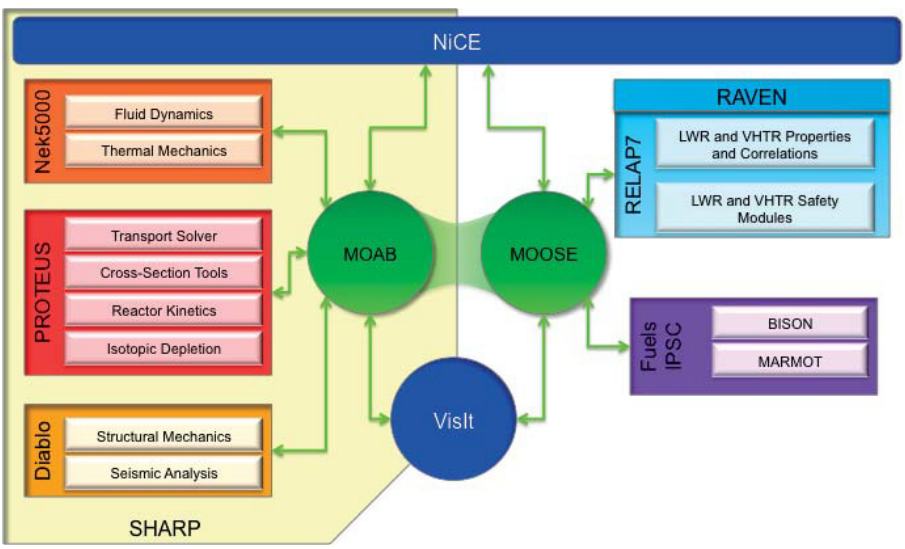


FIG. 1. Schematic view of NEAMS RPL components and connectivity.

3.1.1. High fidelity cross-section module

The high fidelity cross-section module, based on the code MC^2-3 from the PROTEUS suite, provides tools for generation of ultrafine group cross-section libraries for fast reactor neutronics analyses.

3.1.2. Subgroup cross-section module

The subgroup module provides the tools needed to generate both cross-section libraries suitable for transport codes using the subgroup method and libraries that can be used to implement the subgroup method in transport codes, including the high fidelity transport component of the NEAMS neutronics module. This method provides a lower computational cost alternative to ultrafine cross-section libraries generated by the high fidelity cross-section module while retaining reasonable accuracy.

3.1.3. High fidelity transport module

The high fidelity transport module, based on the code UNIC [9] from the PROTEUS suite, provides tools for analysis of neutron transport phenomena within the reactor core. The module is, in essence, multiscale, because traditional homogenization approaches may be used to reduce computational cost — at the expense of accuracy. The module will eventually offer a variety of transport solver options, but initial development efforts will focus on a second order, non-conformal, unstructured finite element discrete ordinates method. The module will be able to make use of both the high fidelity and subgroup cross-section modules.

3.1.4. Near term kinetics module

The near term kinetics module, based on existing code within the nodal diffusion method code DIF3D-K/VARIANT, provides the functionality needed to complete near term transient reactor analyses while longer term development efforts are being executed. Development efforts related to this module are limited to integration of existing code with the NEAMS integrated framework but may be expanded if required by planned assessments of the high fidelity transport module.

3.2. Thermal fluid modules

The NEAMS thermohydraulics tools provide a complete multiscale thermal mechanics and fluid dynamics analysis capability. The one dimensional lumped parameter capabilities of the thermohydraulics tools are based on the system analysis code RELAP-7, and the three dimensional mechanistic capabilities are based on the CFD code Nek5000 [10–13]. The NEAMS thermohydraulics tools include the three major modules described in the sections below.

3.2.1. RELAP-7 system analysis module

The RELAP-7 system analysis module, based on the code RELAP-7, provides one dimensional lumped parameter system performance and safety analysis capability. The module provides a second order finite element implementation of a seven equation transport model. The module relies on conventional engineering correlations to account for multidimensional phenomena in the one dimensional representation. Early development is focused on the capabilities needed for LWR safety analyses and will be completed in collaboration with NE's Light Water Reactor Sustainability Program. The NEAMS Program is developing a limited set of SFR specific extensions to RELAP-7 as a component of the NEAMS RPL.

3.2.2. High fidelity CFD module

The high fidelity CFD module provides predictive, mechanistic simulation of turbulent fluid dynamics and thermomechanics using the highly scalable direct numerical simulation and large eddy simulation capabilities of the spectral element method code Nek5000. In direct numerical simulation, no engineering models are employed to describe the impacts of multidimensional turbulence, and the full Navier-Stokes equation set is solved. In large eddy simulation, the smallest turbulence length scales — those that are much smaller than the length scales of the computational mesh employed — are modelled rather than directly simulated. As a consequence, the computational cost of direct numerical simulations and large eddy simulations is high, and the tools are best used as part of a multiresolution hierarchy in which they serve to inform engineering models used by lower fidelity methods and to aid in benchmarking of lower fidelity simulations. The Nek5000 code is available as a standalone open source module.

3.2.3. Intermediate fidelity CFD module

The intermediate fidelity CFD module provides engineering scale simulations of multidimensional turbulent fluid dynamics and thermomechanics using reduced fidelity Reynolds averaged Navier-Stokes (RANS) methods. In these methods, semi-empirical engineering models are used to describe all turbulence in the system. The primary RANS capability of the NEAMS thermohydraulics tools is implemented within the highly scalable spectral element solver of Nek5000. The module also provides limited connectivity to the commercial CFD code STAR-CCM+ [14], which is used by both nuclear energy R&D organizations and industry. This connectivity allows users to leverage their prior investments in complex CFD models of reactor cores and components.

3.3. NEAMS RPL structural mechanics modules

The NEAMS structural tools provide structural mechanics and material performance analysis. The structural mechanics module, based on the implicit finite element code Diablo, supports engineering scale analysis of structural performance of integrated structures such as fuel assemblies, reactor vessels and containment building. The seismic analysis module extends the capabilities of the NEAMS structural tools with a variety of soil–structure interaction modelling methods. A structural materials module will be developed as an extension of the microstructure models for fuel components provided by the NEAMS fuels product line.

3.4. NEAMS integrated framework modules

The NEAMS integrated framework provides a suite of capabilities for integration of the NEAMS Toolkit physics modules to enable simulation of multiphysics or multiscale phenomena. The framework includes support for integration of physics modules using a unified operator (derived from MOOSE [15]) or a split operator (derived from MOAB [16]). It also includes tools to support management of mesh based and geometry based data and interpolation between mesh distributions or geometry representations used by different physics modules.

3.5. NEAMS meshing modules

The NEAMS meshing tools provide a capability for generation of computational meshes describing reactor geometries that can be used by the physics modules of the NEAMS ToolKit. Current efforts focus on simplification of the process of generation for SFR core components using text files or user interface input. Considerable effort has been invested in parallelizing these modules to improve mesh development times. Support is also provided for a variety of other common mesh formats.

3.6. NEAMS user interface modules

The NEAMS Integrated Computing Environment module [17] provides the user interfaces for problem definition, computer job control, and data analysis and visualization. The environment also provides access to a suite of utilities to support data analysis, parametric studies, workflow management, generation of model documentation and access to code documentation.

4. MAJOR INNOVATIONS AND ACCOMPLISHMENTS

The NEAMS project has adopted a modern code development strategy that relies on highly modular component wise development to provide flexibility in the use and application of the code suite. The NEAMS Toolkit suite applies the principles of object oriented programming and leverages many existing solution and support libraries such as PETSc [18], Trillinos [19], LibMesh [20] and MOAB [16]. The NEAMS project also leverages the MOOSE [15] software development platform, which itself uses PETSc, Trillinos and LibMesh. With many internal and external dependencies, more traditional approaches to software verification, which often rely on line-by-line reviews completed by secondary reviewers at the end of the development cycle, are difficult to implement and often less reliable than desired. The project has adopted software quality assurance practices that build upon a foundation of rigorous version control and tracking, automated verification and automated documentation.

One common thread in the difficult use cases that have been identified is the importance of geometry, or small changes in geometry, to the prediction of multiphysics behaviours. To address this challenge, the toolkit enables generation of computational models that represent the geometry of the system with unprecedented fidelity. Unstructured computational mesh approaches are becoming increasingly common in engineering analysis software packages to enable more realistic representations of component geometry. The NEAMS RPL leverages these approaches and includes a unique neutron transport solution capability [21, 22] that can use both these fully unstructured computational meshes for more accurate representation of core geometry and high resolution ultrafine group nuclear cross-section data where they are available. This important feature enables the toolkit to provide more accurate predictions of local reaction rates in fuel and structural material regions and, more precisely, assess reactivity coefficients, especially those related to core component distortions. Initial demonstrations have focused on the application of the toolkit to well described benchmark problems with complex geometries, such as the Zero Power Reactor-6 Assembly 7 (ZPR-6/7) experiments [23] shown in Fig. 2, and the core of the Advanced Test Reactor (ATR) [24].

The NEAMS Toolkit also enables the application of high fidelity CFD and structural mechanics analysis to advanced reactor performance and safety assessments. These simulation tools provide insight into multidimensional phenomena that cannot be easily evaluated using conventional correlation based methods. The high fidelity capabilities of the toolkit enable accurate assessment of localized temperature, flow and material stress effects resulting from component contact, flow stagnation, flow stratification and structural deformation. Evaluation of relevant phenomena in reactor components using these tools requires access to significant computational resources with hundreds of thousands of CPU cores. Therefore, they have been adopted as part of the NEAMS multiscale, multiresolution strategy and are expected to provide benchmarking or calibration data for lower fidelity methods that are part of the NEAMS RPL. Initial demonstrations of these capabilities have focused on separate effects validation exercises in which thermal stratification or striping is observed [25, 26], and a thorough assessment of the evolution of flow fields in wire wrapped SFR fuel assemblies of various configurations [27–31], as shown in Fig. 3.

As the physics analysis tools rely on unstructured computational mesh descriptions, generation of these meshes and management of data associated with them are critical to the success of the package. The reactor geometry mesh generation tool, called MeshKit, has been developed as part of the NEAMS RPL. MeshKit reduces mesh development time for full core geometries, without spacer components, from days to minutes [33]. The tools have been demonstrated for SFRs, PMRs, LWRs and complex test reactor cores such as the ATR.

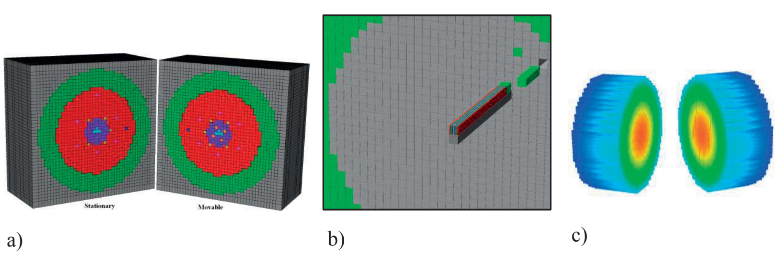


FIG. 2. (a) Geometry of the ZPR-6/7 critical experiment, (b) detailed geometry of heterogeneous fuel drawer geometry from simulation and (c) predicted flux in thin fuel plate at centre of each drawer.

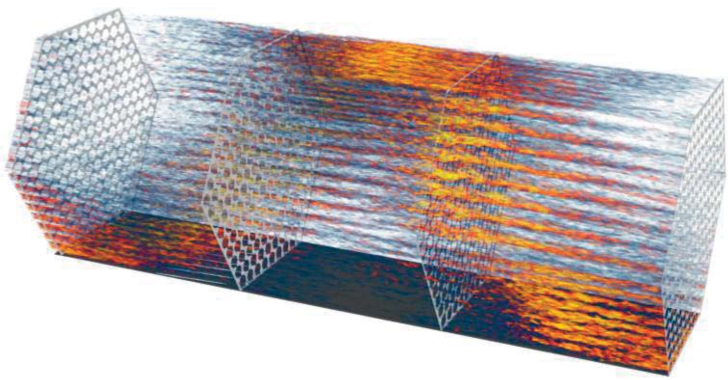
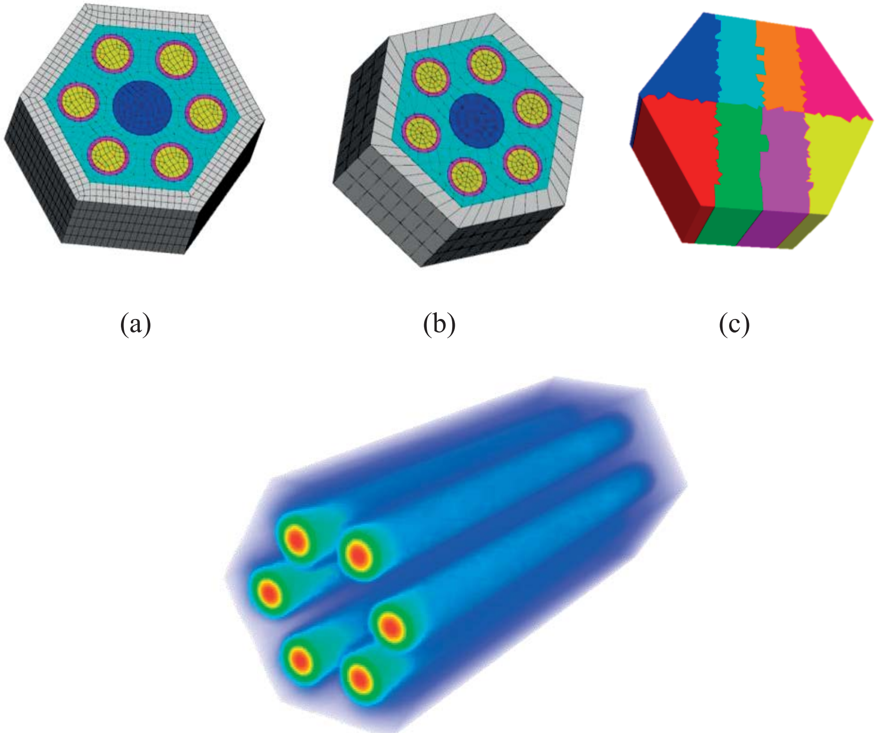


FIG. 3. Predicted evolution of turbulence in a Nek5000 spectral filtered large eddy simulation of a 217 pin wire wrapped fuel assembly using n = 1.01 billion grid points [32].

For the NEAMS RPL, one of the most significant challenges lies in managing the large mesh based data sets generated by each of the high fidelity physics modules. The data stored on the mesh must be available to the single physics application from whence it came as well as to the other physics modules that are integrated to complete a multiphysics simulation. Perhaps the most important innovation of the NEAMS RPL is a powerful suite of mesh based data management and code integration tools that enable the integrated multiphysics modules to move beyond simply sharing cell centred data from the previous iteration, as is typically done in recent coupled code demonstrations. The code integration framework provided by the MOAB toolset enables the data exchange functions to take advantage of higher order information in the solution and MOAB's awareness of the underlying geometry when transferring data between modules using different computational meshes. The goal of this development effort is to significantly reduce the error associated with translation of data between meshes. An initial demonstration of MOAB enabled data exchange among the primary physics modules of the NEAMS RPL toolkit was recently completed for a simplified SFR fuel assembly geometry, as shown in Fig. 4, and more prototypical demonstrations of this capability are in progress.

The NEAMS RPL has adopted a validation hierarchy that addresses the complexity of multiphysics integration. Individual physics modules are subjected to single phenomena unit tests and multiphenomena single physics benchmark tests. Integrated multiphysics modules are subjected to integrated multiphysics tests that represent subsystem and full system behaviours. While existing experimental databases may be adequate for integral system validation exercises,



(a) (b) (c)

(d)

FIG. 4. (a) CFD mesh, (b) neutronics mesh, (c) decomposition and (d) thermal field solution for simplified SFR fuel assembly multiphysics simulation using the NEAMS RPL.

experimental data used in unit and benchmark tests must resolve phenomena at the same length and timescales as the simulations. As an example, validation of CFD simulations requires adequate instrumentation to resolve turbulent flow structures within the test section and the effects of turbulent fluctuations at the boundaries. These new experiments must also provide detailed assessments of many error sources, including instrument error, repeatability error, environmental biases and user biases. The NEAMS RPL supports a pilot project to establish benchmark experiment instrumentation and error assessment requirements [34–36].

5. SUMMARY AND CONCLUSIONS

The NEAMS RPL is an integrated suite of tools that enables high fidelity multiphysics, multiscale simulations for the assessment of performance and safety characteristics of advanced nuclear reactor concepts. The initial release of the fully featured toolkit to the user community, planned for 2018, will support analysis of a variety of advanced reactor types and conditions. However, the initial validation efforts centre on a single challenge — the unprotected loss of flow transient in an SFR. Initial demonstrations of capability for this challenge have already been completed and more prototypical demonstrations are in progress. Some components, including the Nek5000 CFD module, the MOAB framework module and the NEAMS integrated computing environment user interface module, have already been released as standalone open source modules for use by the community at large.

ACKNOWLEDGEMENTS

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NEA ACTIVITIES IN PRESERVING, EVALUATING AND APPLYING DATA FROM FAST REACTOR EXPERIMENTS

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Abstract

The goal of the OECD Nuclear Energy Agency (NEA) in the area of nuclear science is to help member countries identify, collate, develop and disseminate the basic scientific and technical knowledge required to ensure safe and reliable operation of current nuclear systems and to develop next generation technologies. Within these general goals, the current nuclear science programme has three key objectives: (i) to help advance the existing scientific knowledge needed to enhance the performance and safety of current nuclear systems, (ii) to contribute to building a solid scientific and technical basis for the development of future generation nuclear systems and (iii) to support the preservation of essential knowledge in the field of nuclear science. As part of the second and third of these objectives, an extensive programme of work to preserve and evaluate data from integral experiments has been established, including reactor physics, shielding and criticality safety experiments on fast reactor systems. 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 NEA Data Bank international databases. This information is used extensively by the international nuclear science community to validate their modelling and simulation tools. The process can be viewed as part of a broader knowledge management function, where information is gathered, evaluated, linked and made accessible to a wide range of users. The presentation gives details of the main databases maintained and developed by the NEA, focusing on those related to fast reactor applications. The status of recent preservation activities for fast reactor archives in the United Kingdom is also reported, along with an overview of some other NEA nuclear science activities related to fast reactors in the fuel cycle, materials and nuclear data areas.

1. INTRODUCTION

The Nuclear Energy Agency (NEA) is a specialized agency within the Organization for Economic Co-operation and Development (OECD), an intergovernmental organization of industrialized countries based in Paris, France.

The NEA mission is to assist its member countries in maintaining and further developing, through international cooperation, the scientific, technological and legal bases required for a safe, environmentally friendly and economical use of nuclear energy for peaceful purposes. It also aims to provide authoritative assessments and to forge common understandings on key issues as input to government decisions on nuclear energy policy and to broader OECD policy analyses in areas such as energy and sustainable development. In order to achieve this, the NEA works as a forum for sharing information and experience and promoting international cooperation; a centre of excellence which helps member countries to pool and maintain their technical expertise and a vehicle for facilitating policy analyses and developing consensus based on its technical work.

The NEA's current membership consists of 31 countries in Europe, North America and the Asia-Pacific region. Together, they account for approximately 85% of the world's installed nuclear capacity. Nuclear power accounts for almost a quarter of the electricity produced in NEA member countries. The NEA works closely with the IAEA in Vienna — a specialized agency of the United Nations — and with the European Commission in Brussels.

The Nuclear Science Committee (NSC) is one of six standing committees within the NEA and is charged with helping member countries identify, collate, develop and disseminate the basic scientific and technical knowledge required to ensure safe and reliable operation of nuclear systems.

Within these general goals, the current nuclear science programme has three key objectives:

(i) To help advance the existing scientific knowledge needed to enhance the performance and safety of current nuclear systems;
(ii) To contribute to building a solid scientific and technical basis for the development of future generation nuclear systems;
(iii) To support the preservation of essential knowledge in the field of nuclear science.

As part of the second and third of these objectives, an extensive programme of work to preserve and evaluate data from integral experiments has been established, including reactor physics, shielding and criticality safety experiments on fast reactor systems.

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 NEA Data Bank international databases. This information is used extensively by the international nuclear science community to validate their modelling and simulation tools.

The process can be viewed as part of a broader knowledge management function, where information is gathered, evaluated, linked and made accessible to a wide range of users. The presentation gives details of the main databases maintained and developed by the NEA, focusing on those related to fast reactor applications.

The status of recent preservation activities for fast reactor archives in the United Kingdom is also reported.

2. KNOWLEDGE MANAGEMENT AT THE NEA AND THE PRESERVATION OF ESSENTIAL INFORMATION

2.1. Introduction

Since the beginning of the nuclear power industry, numerous experiments concerned with nuclear energy and technology have been performed at different research laboratories worldwide. These experiments required a large investment in terms of infrastructure, expertise and cost; however, many were performed without a high degree of attention to the archiving of results for future use. Given the cost of these experiments and the reduced numbers of facilities and experimentalists now available, it is unlikely that many of these experiments will be repeated in the future.

In response to this situation, the NEA and other national organizations initiated programmes during the 1990s designed to identify, evaluate and preserve essential experimental data and information. These activities can also be seen as part of a knowledge management process where the information is made available to a broad community of scientists and engineers, technical experts who apply the information to their activties, provide feedback to the evaluation/preservation programmes and thereby help to further develop the knowledge base. In addition, by including younger engineers/scientists in the evaluation process, tacit, as well as explicit, knowledge can be transmitted to the 'new generation'.

The NEA Data Bank maintains and distributes several databases of integral experiments for application in the areas of criticality, reactor physics shielding, fuel performance and waste management. The identification and evaluation of the experiments is carried out by technical experts as part of the mandated activities of various nuclear science expert groups. 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).

2.2. Integral experiments databases: ICSBEP and IRPhE

The purpose of the ICSBEP and the IRPhEP is to provide extensively peer reviewed integral benchmark data that can be used by the international nuclear data community for the testing and improvement of nuclear data files and by the international reactor physics, criticality safety, and mathematics and computation communities for validation of analytical methodologies used for reactor physics, fuel cycle and nuclear facility safety analysis and design, and advanced modelling and simulation efforts.

The Criticality Safety Benchmark Evaluation Project was initiated in October 1992 by the United States Department of Energy. The project quickly became an international effort as scientists from other other interested countries became involved. The ICSBEP became an official activity of the NEA's in 1995.

The IRPhEP was initiated, as a pilot activity, in 1999 by the NEA's NSC. The project was endorsed as an official activity of the NSC in June 2003.

A combined total of twenty-four counties have contributed to these two projects, 20 in the ICSBEP and 19 in the IRPhEP. Contributors are: Argentina, Belgium, Brazil, Canada, China, the Czech Republic, France, Germany, Hungary, India, Israel, Italy, Japan, Kazakhstan, the Republic of Korea, Poland, the Russian Federation, Serbia, Slovenia, Spain, Sweden, Switzerland, the United Kingdom, and the United States of America.

The evaluation process entails the following steps: (i) identification of experimental reactor physics related data, (ii) verification of data, to the extent possible, by reviewing original and subsequently revised documentation and by talking with experimenters or individuals who were associated with the experiments or the experimental facility, (iii) evaluation of the data and quantification of overall uncertainties through various types of sensitivity/uncertainty analysis, (iv) compilation of the data into a standardized format, (v) performance of sample calculations for each experiment with standardized reactor physics neutronics codes and (vi) formal documentation of the work into a single source of verified and extensively peer reviewed benchmark reactor physics data.

While coordination and administration of the ICSBEP and IRPhEP take place at an international level, each participating country is responsible for the administration, technical direction and priorities of the project within their respective countries.

2.2.1. ICSBEP Handbook

The September 2012 edition of the ICSBEP Handbook [1] is available on DVD or on the Internet. The DVD version and/or online access can be requested from the ICSBEP Internet sites at http://icsbep.inl.gov or http://www.oecd-nea.org/science/wpncs/icsbep/.

The 2012 edition of the ICSBEP Handbook includes benchmark specifications for the following:

- 723 Pu experiments, of which 118 are metal (111 fast, 4 intermediate, 2 thermal, and 1 mixed), 569 solution (thermal), and 36 compound (7 fast, 4 intermediate, 17 thermal, and 8 mixed);
- 1412 highly enriched U experiments, of which 580 are metal (390 fast, 15 intermediate, 137 thermal, and 38 mixed), 536 solution (3 intermediate and 533 thermal), 289 compound (10 fast, 14 intermediate, 218 thermal, and 47 mixed), 5 mixed metal/solution (thermal), and 2 compound/solution (thermal);
- 61 intermediate and mixed enrichment U experiments, of which 41 are metal (38 fast, and 3 intermediate), 64 solution (thermal), and 156 compound (2 fast, 18 intermediate, 116 thermal, and 20 mixed);
- 1545 low enrichment U experiments, of which 87 are metal (thermal), 117 solution (thermal), 1281 compound (1 fast, 1275 thermal, and 5 mixed), and 60 mixed compound/solution (thermal);
- 244 ^{233}U experiments, of which 11 are metal (10 fast, 1 thermal), 227 solution (190 thermal, 29 intermediate, and 8 mixed), and 6 compound (thermal);
- 503 mixed Pu-U experiments, of which 53 are metal (48 fast, 4 intermediate, and 1 mixed), 72 solution (thermal), and 301 compound (7 fast, 3 intermediate, 274 thermal, and 17 mixed), 56 mixed compound/solution systems (thermal), and 21 mixed metal/compound (13 fast and 8 mixed);
- 20 special isotope experiments, all of which are metal (fast) (^{244}Cm, ^{238}Pu, ^{237}Np, and ^{242}Pu);
- 6 criticality alarm/shielding benchmarks containing 24 configurations with numerous dose points;
- 6 fundamental physics benchmarks, which includes 155 fission rate and transmission measurements and reaction rate ratios for 45 different materials.

2.2.2. *IRPhEP Handbook*

The March 2013 edition of the IRPhEP Handbook [2] is available only on DVD. The DVD version can be requested from the IRPhEP Internet sites at http://irphep.inl.gov or http://www.oecd-nea.org/science/wprs/irphe/irphe-handbook.

The IRPhEP Handbook contains data and, in most cases, benchmark specifications for 129 experimental series from 46 different reactor facilities. Included are the following reactors or assemblies that simulate certain reactor characteristics:

- 6 PWR: CREOLE/EOLE, DIMPLE, DUKE Power, OTTOHAHN, SCCR, VENUS;
- 3 WWER: P-Facility, ZR-6, LR-0;
- 0 BWR;
- 9 LMFR: BFS-1, BFS-2, BR-2, FFTF, JOYO, SNEAK, ZEBRA, ZPPR, ZPR;
- 5 GCR: ASTRA, HTR-10, HTTR, PROTEUS, VHTRC;
- 0 GCFR;
- 5 LWR: CROCUS, DIMPLE, IPEN(MB01), KRITZ, TCA;
- 3 HWR: DCA, ETA, ZED2;
- 0 MSR;
- 1 RBMK: RBMK(CF);
- 5 SPACE: SCCA, TOPAZ, UKS1M, ZPR, ZPPR;
- 16 FUND: ATR, BFS-1, BFS-2, CORAL-1, FRO, HECTOR, IGR, LAMPRE, NRAD, PBF, RA-6, RB, RHF, TRIGA, ZPR, ZEBRA.

3. SECURING THE UK FAST REACTOR ARCHIVE

3.1. Introduction

The Fast Reactor Programme funded by the UK Government ran from 1946 to 1994. During that period it consumed about 50 000 professional person-years of work. The Government ceased funding fast reactor development work in 1993 and reactor operation in 1994, stating that the technology had been brought to the point where further development was not a national responsibility but should be undertaken on a commercial basis by the nuclear industry.

The cessation of government funding meant that the teams that had been engaged in R&D work were dispersed, many staff taking early retirement or moving to work in other fields. Some were redeployed on work associated with the eventual decommissioning of the fast reactor facilities, laboratories, power and research reactors and the reprocessing plant. Efforts were made to create a fast feactor archive as the programme was running to a close, but the work could not be sustained after funding had ceased. At the beginning of 2012, the NEA initiated an investigation of what needed to be done and what could be done to make UK fast reactor data available to the benefit of designers and assessors of future fast reactor systems.

The work was organized under the following headings:

- Determination of the nature of the data generated by the UK Fast Reactor Programme and its state of preservation;
- Preparation of a plan for its retrieval and preservation;
- If valuable archived material is considered to be in a vulnerable location, make arrangements to bring to a 'safe haven';
- Preparation of a report on the UK Fast Reactor fuel programme, including its supporting data.

Much of the work described here was undertaken by C.V. Gregory, formerly Director for Fast Reactors, UKAEA.

3.2. The UK Fast Reactor Programme

The aim of the original UK Fast Reactor Programme was the design and operation of a prototype fast reactor power plant from which a series of commercial power plants would be developed. In reality, the end point of the programme was the construction and operation of the 250 MW(e) Prototype Fast Reactor (PFR) at Dounreay, which operated for 20 years between 1974 and 1994. The technological, design and operational expertise thus gained provided the UK's contribution to the design and development of the European Fast Reactor (EFR), a project that ran between 1988 and 1993.

Underpinning the PFR project was a major R&D programme, starting in 1946 and leading, through the construction of critical facilities and research reactors providing core physics data, to the Dounreay Fast Reactor (DFR), a research power reactor, first critical in 1960. The DFR's principal contribution was a huge amount of research information on the behaviour of fuel and core materials in normal and extreme conditions, a programme that was continued in the PFR and Dounreay's PIE facilities until the early 1990s. Note that it was in the DFR that the important phenomenon of neutron induced voidage in fuel and structural steels was first identified.

The UK's fast reactors were cooled by liquid metal; in the DFR a sodium/potassium eutectic alloy was used, and in the PFR sodium. Major research programmes were undertaken to understand the characteristics of these unusual coolants both within the reactor and without. Thus, chemistry and metallurgy in the liquid metal environment in the reactor, secondary circuits and steam generators were prime topics, as was the interaction of water/steam and sodium in the event of steam generator leakage. In this area, the UK programme was very much more advanced than that in other countries owing to the early decision to use realistically commercial designs of steam generator in the PFR.

The fast reactor cannot be considered without its associated fuel cycle. The technology of reprocessing of the fuel is very different to that in the thermal reactor fuel cycle in that the enrichment, radiation and heat loads are of a different and more challenging order. The UK project was unique in that it included its own reprocessing plant, initially for fuel from the DFR and later, after modification, for PFR fuel. Other countries with fast reactor programmes either chose not to reprocess, relying instead on highly enriched uranium to fuel their reactors, or reprocessed in existing facilities by blending fast reactor with thermal reactor fuel. Thus, the experience of reprocessing and the underlying technology gained from the British programme is of particular significance.

3.3. Sources of information

It was necessary first to establish the location of the UK fast reactor information. The following potential sources were searched:

- The UK National Archives located at Kew in London, where all formal reports and the papers of the various fast reactor working groups have been archived. Material that is more than 30 years old is publicly available. The remainder can be obtained under the Freedom of Information Act. Thus, material created between 1946 and 1982 is readily accessible, while reports and papers produced in the last decade of the programme are less easily available from the National Archives.
- The UKAEA archive at Harwell. Enquiries have indicated that there is very little fast reactor data stored in the Harwell archive.
- British Nuclear Fuels (BNFL), as the manufacturer of fuel assemblies for the PFR, is known to have had a large collection of information on fast reactor fuel. Moreover, BNFL was involved in all the main fuel design and technology working groups and would have kept its own archives of the material issuing from the these. Following the cessation of Government funding for fast reactor R&D, BNFL took on the funding of a small continuation programme for a further 5 years. During this period, BNFL put effort into collecting and safeguarding fast reactor fuel data from the former UKAEA archives.

- National Nuclear Laboratory (NNL). BNFL has ceased to exist as an entity, however, the NNL has inherited the R&D function and its technical archive.
- In the final months of the UKAEA fast reactor project a ‘super archive’ was created at Risley. This archive was bequeathed to AEA (Technology), a successor to UKAEA, later to be privatized. A few years after privatization, AEAT withdrew from nuclear work and the archive was lost. It is understood that those elements of the archive associated with fast reactor fuel technology were taken over by BNFL.
- A number of senior staff from the Fast Reactor R&D Programme kept their private archives when it was realized that no formal system was going to be created. Three such private archives have been amalgamated as a result of the present project.

In the course of the project, it was discovered that North Highland College in Thurso (part of University of the Highlands and Islands) holds a significant collection of old journals (e.g. Journal of the BNES, Annals of Nuclear Engineering, Annual Review of Nuclear Science, Journal of Nuclear Science, Nuclear Safety, etc). These journals contain many fast reactor reports. In addition, there is an archive of all the papers submitted to the public enquiry on the siting at Dounreay of the European Demonstration Reprocessing Plant (EDRP). North Highland College no longer wishes to house the collection. Steps have been taken to catalogue and store these items.

Beyond creating a large collection of references, it is necessary to provide a framework in which to present and order the material. Furthermore, an outline of each of the main technical areas covered by the programme is needed in order to provide the logical and historical sequence of the fast reactor programme, thus providing a context for the collection of references. Without such a context there is the risk that the list will be comprehensible only to those directly involved in the programme. The resulting documents will, in effect, provide a summary of the history of the UK fast reactor project.

The creation of summary documents for each technical area relies mainly on editing combined and existing documents and is not necessarily a hugely laborious task. The first such document covering the fuel development programme has been largely completed.

3.4. Progress with the UK fast reactor references and bibliography

The majority of effort has been directed to uncovering the fast reactor material in the National Archives at Kew. It is not a simple job since there is no classification system that allows one to search under ‘fast reactors’, for example. It is necessary to know the committee and reporting structure of the old UKAEA and to search under the likely working groups and categories of formal reports. Many person-hours have been spent at Kew and this effort has yielded about 1000 relevant references. The cost of photocopying the references is prohibitively high; and current planning is to photograph the material and later to edit it into pdf format.

A search of the index to the NNL archives has yielded a further 1000 titles. This area will be pursued in the next phase. Three private collections of fast reactor material have been amalgamated and stored. The work of cataloguing them is in progress. Also, the North Highland College collection of journals and papers has been taken over by the project and is being catalogued.

4. OTHER NEA NUCLEAR SCIENCE ACTIVITIES RELATED TO FAST REACTORS

4.1. Introduction

The NEA is involved in a number of activities related to fast breeder reactors covering both strategic and scientific issues. These are conducted under the guidance of the Nuclear Development Committee (NDC) and the NSC. Work in the science area is carried out by several expert groups, notably those organized by the Working Party of Scientific Issues of the Fuel Cycle, which deals with issues arising from various existing and advanced nuclear fuel cycles, including fuel cycle scenarios, separation chemistry and flowsheets, innovative fuels and materials, and waste management. The Working Party of Scientific Issues of Reactor Systems and the Working Party of Scientific Issues of Multi-scale Modelling of Fuels and Structural Materials for Nuclear Systems are also contributing.

In addition, an Expert Group on Integral Experiments for Minor Actinide Management (EG on IEMAM) is in the process of completing its study on the availability and requirements for this this type of modelling activity.

4.2. Expert Group on Integral Experiments for Minor Actinide Management

The detailed design of transmutation systems with reliable accuracy and the precise prediction of the composition of the spent fuel are challenging since the quality of the nuclear data for minor actinide (MA) nuclides is variable and there is a lack of integral experiments against which models can be validated. In contrast, it should be remembered that nuclear data for the major actinides, such as ^{235}U, ^{238}U and ^{239}Pu, have been improved over many years, based on a large number of differential experiments, and validated against a broad set of integral experiments using accelerators, critical facilities and experimental reactors. The integral experiments on MAs, however, may be more difficult than those on the major actinides due to several problems/challenges, for example, restrictions at facilities, difficulty of sample preparation, influence of background radiation on measurement techniques, etc.

In response to these challenges, the NEA's NSC established the EG on IEMAM in 2009.

The objectives of the EG on IEMAM are to review integral experiments for validating MA nuclear data, to recommend additional integral experiments and to propose an international framework to facilitate them from the viewpoints of the MA management. The members discussed the following subjects: (i) requirement of nuclear data for MA management (including evaluation of target accuracy, comparison of uncertainty analysis results among nuclear data libraries and identification of important nuclear data), (ii) reviewing existing integral data and identifying specification of missing experimental work to be required, (iii) identifying the bottlenecks, such as availability of MA sample and experimental facilities, and considering possible solutions to them and (iv) proposal of action programme for international cooperation.

The main outcomes with respect to integral experiments are described below.

4.2.1. Reviewing existing integral data

The members of the EG on IEMAM brought and reviewed information about the existing integral experiments related to the MA management. After reviewing, the experiments were classified into four groups: (i) basic experiments using critical facility, (ii) sample irradiation experiments using a reactor, (iii) mock-up experiments and (iv) accelerator–reactor experiments. Some of the key findings are summarized in the following sections.

4.2.1.1. Basic experiments using critical facility

Comparison with the basic experimental data obtained at critical facilities and calculation results has been effective for the validation of the neutron cross-sections. Information on experiments performed at 14 critical facilities of 7 countries includes (a) reaction rate ratio measurements, (b) small sample reactivity worth measurements and (c) criticality measurements.

4.2.1.2. Sample irradiation experiments

Sample irradiation experiments performed in 9 reactors of 6 countries were identified as being of direct application to MA modelling validation.

Irradiation experiments are very useful in evaluating cross-section data and in validating transmutation rates for MA. To improve the cross-section data used for predictions of transmutation rates, the samples are irradiated in a variety of neutron spectra. Analysis of the experiment needs detailed information on irradiation conditions, such as reactor power history, neutron spectrum at the irradiation point, cooling time, etc. Some of the experiments, particularly those carried out prior to the 1990s were found to be missing some of these data.

4.2.1.3. Mock-up experiments

There is only one series of partial mock-up experiments for an MA loaded core, carried out at the BFS facility of IPPE in the Russian Federation. About 10 kg of ^{237}Np dioxide pellets were loaded in the central core region and the core characteristics (e.g. criticality, reactivity coefficients) were measured.

A calculation study showed that the mock-up experiments with loading massive MA fuels were very effective in reducing the uncertainty of the core characteristics in the design of full scale reactor systems. The EG on IEMAM concludes that more mock-up experiments would be of great benefit.

4.2.1.4. Accelerator–reactor experiments

Several experimental studies utilizing subcritical accelerator driven systems (ADS) have been carried out. In Europe, the MUSE experiments were carried out in the MASRUCA facility and the GUINEVERE experiments have been made in the VENUS facility. The purposes of these experiments was to investigate the validation of the subcriticality monitoring for an ADS and the applicability of conventional calculation systems for the fast reactor. In Japan, ADS experiments have been conducted in a well thermalized critical assembly (KUCA A-core) in order to evaluate the basic characteristics of ADS by 14 MeV neutrons generated by D-T reactions or spallation neutron generated by the FFAG proton accelerator.

4.3. Overview of fast feactor related activities in the Working Party on the Scientific Issues of the Fuel Cycle and the Working Party of Scientific Issues of Multi-scale Modelling of Fuels and Structural Materials for Nuclear Systems

4.3.1. Benchmark study on scenario codes

Several scenario codes have been developed to study the future of nuclear energy in different countries. These codes simulate scenarios for nuclear energy at national, regional and worldwide levels. Existing scenario codes developed in Belgium, Canada, France, Japan, Spain and the USA were compared in terms of capabilities, modelling and results, and a benchmark between the different codes was established. Firstly, the benchmark involved depletion calculations for Generation-II and Generation-IV calculations. Comparison of various scenario codes applied to three different scenarios (open fuel cycle with direct disposal of spent fuel, single recycling of plutonium in LWR and transition between a Generation-II LWR fleet and a Generation-IV fast reactor fleet recycling Pu and MA) was then studied.

The results and analysis lead to the conclusion that there is good coherence between the codes for the depletion scenario. However, differences were observed in the dynamic scenario partly due to different physical models and different levels of modelling flexibility, especially for fast reactor scenarios.

4.3.2. Innovative fuels and modelling methods

Comparative studies are being undertaken to support the development of innovative fuels (including innovative clad materials) that can be implemented in advanced nuclear fuel cycles. The fuel types of interest are those that contain MA as opposed to standard fuels (i.e. uranium or uranium-plutonium fuels that are currently being used in the fuel cycle), especially oxide, nitride, metallic and dispersion (CERCER and CERMET) fuels. Special mechanical forms (e.g. particle fuels, vibropac and sphere-pac fuels) are also considered. Comparisons for each fuel form include the fabrication processes and irradiation performance of the fuels along with the available fundamental properties and characterization activities. A state of the art report on innovative fuel concepts is being prepared. In parallel, modelling studies are being conducted, comparing current fuels and innovative fuel designs. Multi-scale modelling methods will help describe the phenomena induced by irradiation in structural nuclear materials of current and future reactors.

4.4. Modelling of transients in sodium cooled fast reactors (SFR Task Force)

In the frame of the Working Party on Reactor and System, an international task force has been established to work towards a shared neutronic analysis of several Generation-IV SFR concepts. Details of the work to date are given in another paper at this conference.

The objective is to design cores that can passively avoid damage when the control rods fail to scram in response to postulated accident initiators (e.g. inadvertent reactivity insertion or loss of coolant flow). The analysis of such unprotected transients depends primarily on the physical properties of the fuel and the reactivity feedback coefficients of the core. The work consists of the following four steps:

(i) Compile a 'state of the art' report: review past and recent studies performed in the framework of SFRs and build a bibliographic repository which would stress core transient behaviours as a function of fuel characteristics (oxide, carbide, nitride and metal).
(ii) Perform a parametric study based on two different core sizes (3600 MW(th) and 1000 MW(th)) and three fuel types (oxide, carbide and metal) for identifying the advantages and drawbacks of each concept.
(iii) Based on the results obtained in the previous step, transient calculations will be performed on a few selected cases for principal unprotected transients.
(iv) Synthesis of the whole work into a final report, including recommendations to improve safety and future work towards severe accidents and MA management.

So far, results have been obtained mainly from analyses at Argonne National Laboratory in the USA and at the Commissariat à l'Énergie Atomique et aux Énergies Alternatives in France. These early results show that for the different core concepts analysed, a satisfactory agreement between participants was obtained despite the different schemes of calculation used. A good agreement is generally obtained when comparing the burnup composition evolution, the delayed neutron fraction, the Doppler coefficient and the sodium void worth. However, some noticeable discrepancies between the K-effective values were observed and are explained in this paper. These are mostly due to the different neutronic libraries employed (JEFF3.1 or ENDF/B-VII.0). Plutonium isotopes, fission cross-sections are responsible for a large part together with the sodium inelastic cross-section.

5. CONCLUSIONS

The NSC is one of six standing committees within the NEA and is charged with helping member countries identify, collate, develop and disseminate the basic scientific and technical knowledge required to ensure safe and reliable operation of current nuclear systems and to develop next generation technologies.

As part of meeting these objectives, an extensive programme of work to preserve and evaluate data from integral experiments has been established since the mid-1990s, which includes information from reactor physics, shielding and criticality safety experiments on fast reactor systems. The NEA Data Bank maintains and distributes several databases of these integral experiments, notably through the ICSBEP and IRPhE projects.

More recently, programmes of work have been established to help preserve data from the UK Fast Reactor Programme and from various experiments related to MA management.

The data obtained from these programmes are made available to the nuclear science community to provide high quality benchmarks against which modelling methods can be validated. In addition, the involvement of younger scientists and engineers to work alongside well established experts in the process of evaluating the information is a highly efficient means of transmitting tacit knowledge to the new generation of nuclear specialists.

ACKNOWLEDGEMENTS

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FAST REACTOR DEPLOYMENT, SCENARIOS AND ECONOMICS

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A GLOBAL ASSESSMENT OF FAST REACTORS IN THE FUTURE

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Abstract

Various criteria will be presented and used for assessing the future of sodium cooled fast reactors (SFRs) on a worldwide basis, including sustainability, economics, contribution to maintaining nuclear R&D excellence, long term acceptability of nuclear energy, leading position in nuclear energy industry for countries developing SFRs and diversification of the risks and insurance. One of the main concerns is public acceptance, which may vary over time for a number of reasons. If it is assumed that safety and non-proliferation concerns will be dealt with effectively, acceptance will most probably be obtained and the question will not be whether to launch SFRs on an industrial scale, but when and where. An assessment of the market will also be provided in this paper. The world market for industrial Gen IV SFRs is expected to be between 0 and 2 units (1500 MW(e)) per year based on an optimistic approach, before economic competitiveness is reached, and 10–15 later. Though there are large uncertainties on the exact period at which economic competitiveness will be reached, it is most probably likely to occur sometime during the second half of the century. In the future, the advantages of SFRs will likely grow significantly faster than any disadvantages.

1. ENERGY STRATEGIES FOR THE FUTURE OF HUMANKIND: NEEDS AND SOLUTIONS

The main institutes and organizations dealing with energy issues (International Energy Agency, World Energy Council) state that the use of nuclear energy will be an essential part of the world's energy balance in this century. In its most recent report, the International Energy Agency (2012) [1] even stated that the expected rate of development of nuclear energy could be too slow for humankind to be able to limit greenhouse gas concentrations (CO_2 here) to 450 ppm by 2100 (for an average rise in temperature potentially limited to 2°C). The development of nuclear energy is forecast to increase, with power likely to double by 2050 and with very few countries having chosen to abandon this type of energy. The Fukushima accident has slowed down the process, but the delay in the development of nuclear energy in the world will be limited to around 5 years [2].

2. HOW TO ASSESS THE LONG TERM SOLUTIONS: A MULTI-CRITERIA GRID

The share of sodium cooled fast reactors (SFRs) in the nuclear mix can be assessed by means of a multi-criteria evaluation of the use of these reactors. Various criteria will be presented and used for assessing the future of SFRs on a worldwide basis:

- Sustainability;
- Economics;
- Contribution to maintaining nuclear R&D excellence and dissemination of the best technologies;
- Long term acceptability of nuclear energy;
- Diversification of the risks and insurance.

2.1. Sustainability

2.1.1. Uranium availability

Natural uranium resources are limited [3], which means that current light water reactor (LWR) based nuclear energy is not sustainable for the whole of the century. The use of fast reactors is therefore essential. Although sustainability is an essential criterion, it is not the only factor to be taken into account, and the following sections analyse these other criteria.

2.1.2. Spent fuel and waste management

As well as the much better use of natural resources, the main advantage of a fast reactor treatment-recycling strategy is to fully use the fast reactor plutonium present in LWR spent fuels.

Fast reactors enable the stocks of plutonium to be managed and their total amount to be adjusted to what is considered to be optimum for a given fleet. They could also decrease this stock, if nuclear energy were halted. There would however be long time constants (several dozen years at least). This is shown in Fig. 1.

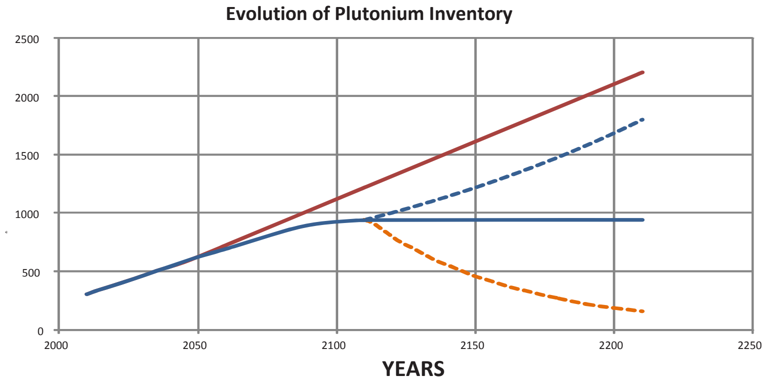

FIG. 1. Possible changes to plutonium stocks as a result of the development of fast reactors in France. Red line: underlying scenario without fast reactors. Blue line: with fast reactors and assuming the power demand is flat (as will probably be the case for France), the Pu stocks will also be flat. Anyway, changes to stocks are possible and are given by the dotted lines: the lower dotted line corresponds to a scenario in which nuclear energy is halted with reduction of the plutonium inventory, and the upper one to a scenario for which the electricity from fast reactors is increasing). Source CEA.

Fast reactors can also provide a broad contribution, via the minor actinide transmutation techniques, when putting a range of solutions in place to deal with radioactive waste in permanent storage.

The benefits of nuclear energy with fast reactors should be felt in terms of quality of Pu confinement (fast reactors need a closed fuel cycle, so almost no plutonium will be present in the final wastes), thermal power and toxicity (if applicable, if the transmutation option is used). Even if today the plutonium were already absent in the vitrified waste packages (continuing with current practices), the various waste transmutation techniques should enable long term radiotoxicity to be limited further by a factor of up to 10.

These types of packages could require a smaller storage area, for example, up to two times less, or even less, according to ANDRA (after 120 years' storage):

- Reduction of the space taken up by modules for high level waste: #9;
- Reduction of the space taken up by high level waste areas (including galleries): #3.5;
- Total reduction for storage (including intermediate level long lived waste): <2.

2.2. Economics

2.2.1. Electricity production cost

The investment cost accounts for the greatest proportion of the cost of electricity, both for current water cooled reactors and future fast reactors. According to current studies, fast reactors will involve additional investment costs compared with water cooled reactors. This additional cost is currently estimated to be about 20–30%. But the electricity produced by a fast reactor could become competitive with that produced by a water cooled reactor via savings in terms of consumption of natural uranium and uranium enrichment services. Calculations show that all that would be needed for these reactors to be competitive would be an average uranium cost per kg of between €250 and €600, if the additional investment cost were that quoted. In the knowledge that uranium had already reached a cost of more than €200/kg on the spot market in 2006–2007, these values could be significantly exceeded within several decades. Also, reactors are chosen based on anticipated costs over the whole of their service life, i.e. probably at least 60 years.

From the time when the costs become balanced, the production cost of fast reactors will hopefully be lower than that of water cooled reactors because uranium is a rare resource and its cost will tend to increase. In addition, if the functions for transmutation of the most long lived toxic radionuclides (minor actinides) were to be chosen, French studies [4] show that the cost per kW·h would increase by less than 9%.

2.2.2. Impact on the balance of trade

The direct impacts of developing fast reactors would affect the following:

- Imports of energy materials;
- Imports/exports of technology, goods and services associated with fast reactors.

With regard to the first: if we take the example of a country for which nuclear electricity production for the period in question is 100 TW·h, based on a uranium price of up to €300/kg in that period[1], the savings on uranium imports would be in the region of 0.5 billion euros a year, in comparison with a reference

[1] This refers to current prices and not anticipated prices over the operating period of the reactors, which will by nature be higher.

for water cooled reactors. Of course, the effects may be much greater if fast reactors take the place of fossil fuel powered units. They would be in billions of euros a year for such a country.

There are also macroeconomic effects associated with investments (which affect the activity, induced imports and prices) and electricity prices (see below). These effects include, for countries developing fast reactor technologies, the dual advantage of only importing a small proportion of the power plants to be built and exporting reactors and reactor cycle services.

2.2.3. Impact on employment

Closely associated with the challenge of creating energy solutions capable of meeting national requirements and capturing international markets is the issue of employment, which, in addition to considerations of security of supply and reduction of greenhouse gas emissions, is the guarantee of an economically successful energy policy.

For a country deciding to opt for fast reactor technology, the main elements affecting employment are as follows:

- The implementation of R&D programmes, which always go hand-in-hand with the development of a new technology, but whose scope will differ widely depending on the strategic choices made by the country.
- The building of a series of reactors will, for the most part, be localizable.
- The added value will be potentially even more localizable in the country than may be the case for current reactors, which have a high cycle cost (with a uranium weight which will increase).

At least qualitatively, the fundamentals of the technology in terms of employment are clearly favourable. They will be all the more so if the country in question has developed its expertise in the technology.

2.3. Maintaining nuclear R&D excellence and dissemination of the best technologies

For countries developing fast reactor technologies, an R&D programme is a powerful way of consolidating nuclear R&D as a whole. Unsurprisingly, it is in countries which are already developing second or third generation reactors that the programmes for fast reactors are located. However, the reverse is not true (as in the United States of America, for example). Indeed, the water cooled reactor industry will largely benefit from fast reactor development.

Thus, in the case of France, one of the essential contributions (provided by any project of the size) of the ASTRID demonstrator, apart from the reason for its creation and its own success, is its ability to inject dynamism into teams, which (as they are currently made up or via the experts in today's teams) could lead to progress in various fields:

- Recommendation of new solutions, even for existing nuclear technologies (safety of water cooled reactors in the post-Fukushima context and beyond, advanced fuels, backend of the cycle, etc.);
- Enhancement of methods, models and standards;
- Contribution to the strengthening of current safety approaches via new questioning.

In addition to this, it is possible to identify two types of effects resulting from research:

- Knock-on effects on suppliers and large companies ordering research programmes;
- 'Spillovers' into other sectors, associated with the dissemination of knowledge generated by this research (desalination, hydrogen production, concentrating solar power plants with sodium coolant, new materials, corrosion, instrumentation, etc.).

2.4. Long term acceptability of nuclear energy

The implementation of Gen IV reactors would not be a neutral event in terms of public opinion. It may even be considered to be only possible if there were sufficient social consensus. Conversely, once their safety advantages have been demonstrated, these reactors could be as acceptable as, or even more acceptable than water cooled reactors. Their inclusion in 'mixed' fleets would also be an important factor for the acceptability of water cooled reactor technology.

2.4.1. Acceptance and sustainability of energy production

The sustainable nature of an energy technology is essentially assessed against the criteria of resource depletion, environmental consequences and health impacts. Fast reactors are particularly well placed in this respect, positioning Gen IV nuclear energy in the same group as renewable energies such as solar or wind power. The challenge to be faced is that of reactor accidents (safety) and waste (for which fast reactors are also advantageous). These two topics are covered in the following sections. Another important feature is the social aspect of energy policies, in particular in terms of costs for the population and its most disadvantaged members. Fast reactors also offer a considerable advantage in the future by guaranteeing very low fluctuations in production costs.

2.4.2. Acceptance and security of supply

On the fringe of the foreign trade balance indicator described previously, is the energy security indicator. The development of a technology that will not consume any imported energy materials[2], and at this stage no identified quantities of strategic metals, is in a remarkable position in this respect, being at least at the same level as solar and wind technologies, for which the proportion of imported capital goods is considerable in many countries (the international leaders are very concentrated).

2.4.3. Acceptance and safety

Fast reactors have been developed together with water cooled reactors, and the safety authorities of countries using this second generation type of reactor have not indicated any associated drawbacks. In France, the ASN requires future fast reactors to have at least the same level of safety as comparable water cooled reactors (of the same generation). To achieve this for SFRs basically means improving the management of the risks of core dewatering or boiling and limiting sodium–water or sodium–air (sodium fire) interactions even further.

Experience from current studies on these reactors has already highlighted potential characteristics of SFRs that are favourable for safety. These include, looking in particular at the French concept, ASTRID:

- The pool type reactor concept with a large vessel and a sodium mass in keeping with this, which provides very high thermal inertia, guaranteeing safety.
- A variation of the concept would be to do away with the sodium–water heat exchangers (gas exchangers), avoiding sodium and water interactions.
- The primary system is not pressurized.
- The margin at boiling is very large (typically 300 degrees).
- It is easy to start up natural circulation of the sodium for cooling.

It is possible to remove decay heat via active or passive devices, which have already been tested in the past. Work is continuing in line with the objectives

[2] Except for certain countries with small amounts of depleted uranium.

indicated earlier. France is targeting a high level of safety for these reactors, a very significant improvement in relation to the second generation. It is involved in this regard in particular with the Generation IV International Forum. The success of these efforts depends both on authorization by the safety authorities and on the level of acceptance by the public, the latter going hand in hand with the former.

2.4.4. Acceptance and proliferation risks

Proliferation resistance is a major requirement which will only become more important as the use of nuclear reactors becomes more widespread.

Future fast reactor concepts, with their associated cycles, will therefore have to prove themselves in terms of proliferation resistance. The use of plutonium is not in itself a drawback; there is a system of safeguards and intrinsic measures, such as the absence of separated plutonium at all stages and help to provide a high level of resistance. Favourable factors include:

- The disappearance of plutonium from LWRs as it is used in fast reactors.
- The removal of the enrichment step from the cycle.
- The very low content of fissile material in waste (due firstly to the closed fuel cycle and improved if transmutation is carried out): the waste packages are not subject to safeguards.
- The concentration of materials at a few international locations or ‘platforms’, which also enables ‘small’ nuclear countries to simplify their material cycles and management, via recycling of the plutonium from water cooled reactors in the fast reactors of countries that are historically committed to recycling.
- The current plans for future fuel cycle facilities are based on an integrated design, combining treatment and recycling on the same site. Such a concept keeps transport to a minimum and enables the amounts of buffer stock of fissile material to be reduced.

2.4.5. Acceptance and long term waste management

The advantages that have been highlighted in terms of waste management are certainly important factors for acceptability [5]. Likewise, the fact that technical progress in terms of the packaging quality, toxicity and thermal condition of the waste has historically involved a recycling approach, which is clearly shown with the use of fast reactors, can be taken into account.

'Mono-recycling' first of all enables the storage area to be considerably reduced. With the transmutation of the minor actinides, a possible option in fast reactors, it is thought that the ability to store packages from the fleet in an area two to four times smaller would be an important advantage. It would certainly make any extension much easier, when the time comes, and with all assurances given to the populations in the areas concerned. This possibility would also enable a search to be carried out for another storage location, if that were the decision, with a greater probability of success.

2.5. Diversification of risks and assurance

Research on fast reactors also has advantages in terms of assurance against the risks resulting from the uncertainties concerning major parameters for the development of power production, such as future prices and costs on various markets.

With this in mind, I-tésé has carried out several studies which focus on determining the moment when it becomes profitable to invest in fast reactors, in view of the possible changes in uranium prices. However, this type of approach does not take account of R&D expenditure in the initial phase.

Aware of these limitations, I-tésé has also directly addressed the question of the initial profitability of R&D, in view of the possible future profitability of a programme (in the microeconomic sense only). The studies referred to demonstrate that for given levels of uncertainty (that are considered to be realistic) regarding uranium prices and the potential additional investment costs involved in fast reactors in comparison with future water cooled reactors, the advantage of the option of having fast breeder technology in order to be able to benefit from it when the time comes will be a few billion euros [6].

2.6. Global multi-criteria assessment

The advantage of using fast reactors lies primarily in their virtually zero consumption of natural uranium, which will ensure sustainability of nuclear energy for several millennia. With this technology, nuclear energy reaches a performance level with respect to natural resources that is comparable with that of renewable energies such as wind or solar power from the point of view of time for human society.

In a context where tensions on climatic risks are running very high, international bodies such as the International Energy Agency are in agreement on stating that the use of nuclear power, an energy with a very low CO_2 content, is essential in order to limit global warming, hopefully to 2°C, by 2100. Well before this date, according to the scenarios that are compatible with this trajectory, all the currently estimated uranium resources will very probably have been allocated (i.e. acquired in advance by reactors that are in service). Fast reactor technology can provide a concrete response to this.

Beyond this, before summarizing the situation of the potential market for fast reactors between now and the end of the century, this paper reviews other scientific, technical, social and industrial criteria regarding the implementation of a fast reactor programme, applying an approach that goes beyond the usual quantitative methods.

It would be worth carrying out a complete multi-criteria analysis of fast reactors, looking at the various technologies and the various regions of the world. The main trends from the huge amount of literature that has been accumulated up to now (and in particular studies by I-tésé at the CEA) enable us to draw up an initial table summarizing the advantages and disadvantages of fast reactors in relation to water cooled reactors. These are given in Table 1.

This table, created for a country that is a future exporter of fast reactor technology, shows very positive results for the technology. Each box represents the comparative advantages of fast reactors versus water cooled reactors, while endeavouring to take into account the expected progress of this family of reactors over the coming decades. If we assume that safety and non-proliferation concerns will be dealt with effectively, acceptance will most probably be obtained and the question will not be if SFRs will be launched on an industrial scale, but when and where.

Therefore, numerous areas that are not usually included in technological evaluations, because they are often poorly understood and in particular difficult to quantify, have been explored. It is therefore important to remain cautious about the broad outlines that are given here. However, it seems possible to draw two conclusions at this stage:

(i) The advantages associated with fast reactors, if they are confirmed, clearly outweigh the disadvantages.
(ii) All in all, these advantages will increase, if these reactors are built on a significant scale, in the long term.

TABLE 1. COMPARISON OF THE RESPECTIVE ADVANTAGES AND DISADVANTAGES OF DEVELOPING FAST REACTORS VERSUS WATER COOLED REACTORS OF THE SAME GENERATION *(evaluations in bold are related to countries that are developing and selling fast reactor technology) (ns: not significant)*

Criteria	Short term	Long term
Sustainability	+	+++
Impact on electricity competitiveness	-	+/++
Impact on balance of trade	ns/+	+/+++
Impact on employment	ns/+	+/++
Technological dissemination (and nuclear spillovers) and contribution to maintaining nuclear R&D excellence	+/+++	+/++
Spin-offs in other areas (and other spillovers)	+/++	Idem LWR
Leading position of nuclear energy industry in countries developing fast reactors	ns/++	ns/+++
Acceptability via sustainability	+	+++
Acceptability via security of supply	+	+++
Acceptability via management of storage (toxicity, area)	++	+++
Acceptability via safety	Idem LWR	Idem LWR
Acceptability via non-proliferation	Idem LWR	+
Diversification of the risk via mastery of the technology	ns/+	+/+++

3. SOME MARKET SCENARIOS FOR THE FUTURE

3.1. Long term changes in the price of natural uranium

The uranium market is a global market. There is currently a uranium spot market which accounts for less than 10% of the trade, with the remainder being acquired via long term contracts.

This approach means that it is not possible to obtain information on the likely cost of uranium (in the world and therefore in France) for 2100–2150. This must be done using other methods. A calculation of the global scenario will be carried out initially, using certain assumptions on the criterion for introducing fast reactors (e.g. on the single aspect of economic competitiveness), on the additional investment cost in comparison with LWRs, on the level of global demand and on natural uranium resources by means of a supply curve.

I-tésé has drawn up a set of supply curves indicating the available tonnage of natural uranium for a given uranium extraction cost. Figure 2 gives these very simplified supply curves that have been drawn up based on the information given by the Red Book and more or less optimistic assumptions on the discovery of new resources.

This scenario can then be used to assess the annual consumption of natural uranium, which gives the annual increase in the cost of natural uranium via the supply curves (it represents the increase in the cost of the resource according to its increasing scarcity, but does not take account of market fluctuations).

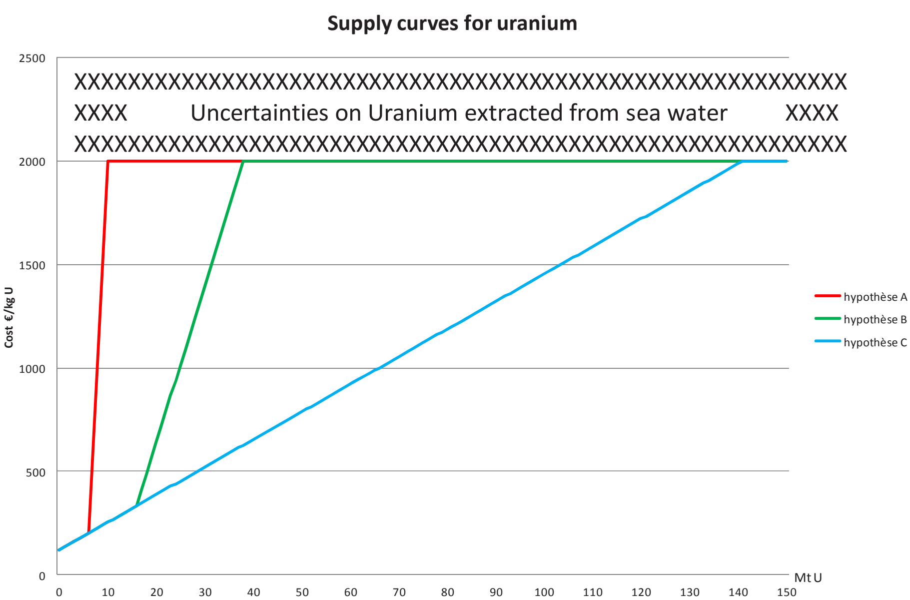

FIG. 2. Uranium supply curves.

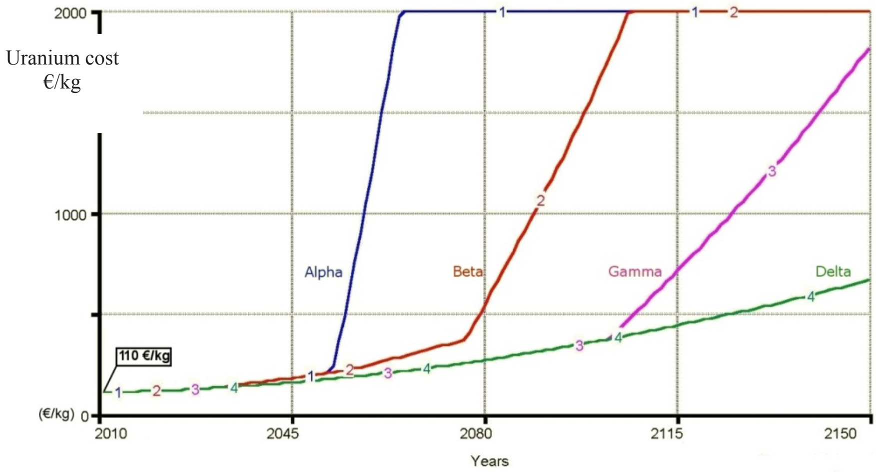

FIG. 3. α, β, γ and δ profiles of the increase in the global cost of uranium (€/kg). The slope increases rapidly when 'cheap' uranium resources are exhausted.

TABLE 2. ECONOMIC COMPETITIVENESS DATES FOR FAST REACTORS *(additional fast reactor investment cost 30%)*

Increase in the global cost of natural uranium	Assumption supply curve	Power demand	Year in which fast reactors become competitive
A	A	High	2040
β	B	High	2070
γ	B	Low	2100
δ	C	Low	2130

Curve *α* (see Fig. 3) corresponds to the use of limited resources (supply curve A) and to a scenario of dynamic development of the global nuclear fleet. This pair of assumptions gives the fastest rise in the cost of uranium and the earliest date for the introduction of fast reactors (see Table 2).

Conversely, curve *δ* corresponds to greater resources (supply curve C) and the scenario with the slowest development of the global fleet (apart from scenarios in which nuclear energy is halted).

The two other profiles, β and γ, correspond to the same supply curve (B), but correspond respectively to scenarios of stronger or weaker development of the global nuclear fleet.

Profile β can therefore be considered to be the most likely given our current knowledge of resources and development of the fleet.

3.2. Consequences on the situation in France assuming economic competitiveness as the only driver

Although France's situation differs from the global situation on two basic points (a much lower growth in power demand and generally enough Pu to avoid needing fast breeders, just iso-generators), the development of its nuclear fleet will depend on the competitiveness of SFRs, which itself depends on the price of uranium on the global market. Consequently, it is not possible to examine scenarios for introducing fast reactors in France without taking account of what is happening in the rest of the world. The following figure indicates what is happening in France for the four curves in Fig. 1 showing the changes in uranium cost.

Although these results may be modified by numerous alternatives on the choice of parameters for the global scenario (changes in the cost of natural uranium, cost of LWRs, additional investment cost or breeding gain of fast reactors, etc.), a number of conclusions can be reached and comments made based on Fig. 4:

- The uncertainties about natural uranium resources and the global development of nuclear energy are such that fast reactors may become competitive between 2040 and 2140, although the most probable time seems to be the second half of this century.
- To determine when fast reactors should become competitive, it is not this year's uranium cost which is involved; the choice of whether to build a fast reactor or an LWR (or another technology) must be analysed over the lifespan of the reactor, which is typically 60 years. It is therefore the change in the uranium cost over 60 years which must be taken into account (incorporating a discount rate). Since this information is not known (except with specific simulations), the competitiveness is calculated using an estimate of the increase in the price of uranium over the lifespan of the reactors.
- To achieve economic competitiveness does not however mean that the fleet will migrate very quickly from LWRs to fast reactors. Figure 4 shows that it will be spread over a minimum of 30 or so years.

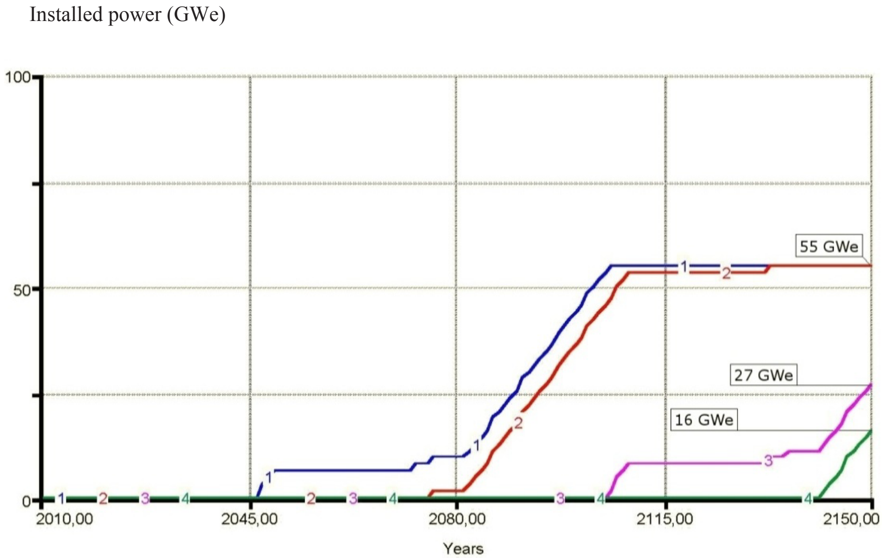

FIG. 4. Results in terms of development of the French fleet of fast reactors: installed net power (GW(e)) for profiles α, β, γ, δ corresponding respectively to curves 1, 2, 3 and 4 (additional fast reactor investment cost 30%).

- However, it may take a great deal longer as two other constraints are also involved, firstly the availability of Pu (which is not a major constraint for France), and then the possibility of replacing reactors as they reach end-of-life, as is the case in France and in countries with a stable or decreasing nuclear fleet. In this case, the age pyramid of the reactors plays an important role. Although fast reactors become competitive when there are not many reactors to be replaced, changing the French LWR fleet to fast reactors may take much longer. This is illustrated by curve α in the Fig. 4, where the change takes 60 years and the fleet becomes all fast reactors at almost the same time as in case β, although the fast reactors do not become competitive until much later.

3.3. The world SFR market

According to the analyses performed by I-tésé, this market can be divided into two phases: before and after ‘microeconomic’ competitiveness. As we have seen, predicting this period of competitiveness is rather complex in light of uncertainties on natural uranium resources together with those on the development of the world LWR fleet. Furthermore, this period of competitiveness will not be the same in every country; it not only depends on the cost of labour, but also on the country's political situation, which is not the same for a country that charges for the processing of spent PWR fuel assemblies for SFRs or LWRs.

Though reaching microeconomic competitiveness will mark a key stage in the development of SFRs, it is clear that the incentive of the first SFR purchasers will be more political than economic. The criteria supporting this first phase of the market will be:

- Safety;
- Energy self-sufficiency of the country;
- Secure energy supply;
- Guarantees relative to the pressure on the natural uranium market;
- Positioning in the high technology industry;
- Plutonium management;
- Waste management;
- Integration of non-proliferation issues.

Therefore, these different advantages can lead some States to make a political decision to build SFRs, including the implementation of suitable governmental funding to counterbalance the excess cost of SFRs, before utilities competing on the market are led to make this decision on the basis of economic criteria alone.

China, India and the Russian Federation are the countries that seem the most capable of funding a 'pre-commercial' SFR to create and develop a market for these reactors. These countries have a solid nuclear industry, more or less extensive experience in SFRs, and, especially, the strong political will to embark on the construction of SFRs before their kW·h cost becomes competitive.

Among the other countries, only a few could instigate the construction of SFRs before they become economically competitive, but on a small scale with the main objective of keeping this technological door open and of taking part in the definition of its standards.

France is planning to build its first industrial scale SFR in 2040, which could be followed by a few more units to ensure the development of an industrial fabric prior to reaching economic competitiveness. If this is reached as early as this decade, SFR investment in France could exceed a dozen reactors over a period of about 15 years (depending on the availability of plutonium resources). This is the most optimistic scenario.

Among the countries capable of building an SFR before 2050, there is Japan which must nonetheless manage the post-Fukushima situation and redefine its own nuclear policy. The Republic of Korea may also be ready within the scope of its nuclear development, but it must first build up its capacity to process LWR fuel while controlling both the technical and political issues associated with this development. The United Kingdom may also move in this direction within the scope of managing its Pu resources. Lastly, the USA had been moving away from a closed cycle for several decades, but it is now gradually backtracking. Even if shale gas prevents nuclear energy from developing, changes remain possible in this very important country, especially if nuclear energy is revived on a global level.

In the first phase — starting not earlier than 2030/2040 — prior to SFR competitiveness, it can be optimistically assumed that about 2 SFRs will be built per year throughout the world. So, the world market for Gen IV SFRs is expected to be between 0 and 2 units (1500 MW(e))/year in the next 2–3 decades.

In the second phase, which should occur sometime in the second half of the 21st century, when SFRs will be deemed economically competitive considering the market expectations, the demand would rise rapidly and between 10 and 15 SFRs can be expected to be built per year throughout the world, depending on the conservative assumptions on nuclear fleets.

Considering the cost and the difficulty of setting up an industrial scale reactor, it seems rather probable that there will be no more than two or three different SFR technologies on the market. Consequently, the number of OEMs will also be limited, which makes it possible to assume that multinational OEM consortiums will be set up.

Even if it is technical possible to start up an SFR with enriched uranium, this solution is economically costly [7] and should not be deployed on a large scale. Once the economic competitiveness of SFRs has been established, the main constraint will be ensuring the availability of plutonium. This issue will limit the number of SFRs that can be built.

It is therefore important to note that the deployment of SFRs will only be possible once a plutonium processing market has been developed. From this viewpoint, the implementation of regional processing centres under the responsibility of the IAEA could be an efficient solution. This solution would also make it possible to recycle — for the benefit of all — plutonium from LWRs in countries which are only just launching a nuclear industry, in SFRs belonging to countries that have already demonstrated control over these technologies.

4. CONCLUSION: A GLOBAL ASSESSEMENT FOR A GLOBAL MARKET

Before summarizing the situation of the potential fast reactor market between now and the end of the century, this paper reviews other scientific, technical, social and industrial criteria regarding the implementation of a fast reactor programme, applying an approach that goes beyond the usual quantitative methods. It seems possible to draw two conclusions at this stage:

(i) The advantages associated with fast reactors, if they are confirmed, clearly outweigh the disadvantages.
(ii) All in all, these advantages will increase, if these reactors are built on a significant scale, preferably in the long term.

Also, it is assumed that if safety and non-proliferation concerns will be treated efficiently, acceptance would be most probably obtained. The dynamics of the world energy demand and the extreme urgency of climate issues are such that, except for a major upheaval (and time constants of the energy sector are known), the question is not in knowing if SFRs will emerge or not, but rather defining when they will develop.

The second part of the paper deals specifically with this point. Taking advantage of this general overview, the world market for industrial Gen IV SFRs is expected to be between 0 and 2 units (1500 MW(e)) per year based on an optimistic approach, before economic competitiveness is reached, and 10 to 15 units later. Though there are large uncertainties in the period at which economic competitiveness will be reached, it is most probably likely to occur sometime in the second half of the century.

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JAPANESE FAST REACTOR DEPLOYMENT SCENARIO STUDY AFTER THE FUKUSHIMA ACCIDENT

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Abstract

As a result of the accident at the Tokyo Electric Power Company Fukushima Dai-ichi nuclear power station (NPS), hereinafter, referred to as the Fukushima Dai-ichi accident, caused by the Great East Japan Earthquake on 11 March 2011, a decision was made to re-examine the strategic energy plan of Japan and the framework of nuclear energy policy. In 2012, scenario evaluations were carried out on options for the nuclear fuel cycle policy according to various shares of nuclear energy supply in the medium term, mainly until 2030 in the Atomic Energy Commission. In parallel to this medium term scenario analysis, the Japan Atomic Energy Agency implemented the long term scenario analyses including fast reactor cycle deployment. As a result, fast reactor cycle deployment brings great benefits to reduction in uranium demand, spent fuel storage, radioactive waste generation and Pu stockpiles, in addition to potential hazard (radiotoxicity) of high level waste in the '20 GW(e) constant after 2030' case. Meanwhile, it was revealed that fast reactor cycle deployment offers benefits to the reduction of radioactive waste generation and Pu stockpiles even when considering the 'gradual decrease from 20 GW(e) after 2030' case.

1. INTRODUCTION

The Japan Atomic Energy Agency (JAEA) has been conducting research and development (R&D) towards commercialization of the fast reactor cycle in cooperation with electric utilities, the Central Research Institute of Electric Power Industry and vendors since July 1999 in order to present an appropriate concept of commercialization of the fast reactor cycle and construct the required system of technologies. Subsequently, as a result of the Fukushima Dai-ichi accident, the Fast Reactor Cycle System Technology Development Project (known as FaCT) phase II [1], which was scheduled to commence in Japanese financial year 2011, is frozen at the moment. Currently, studies are under way that utilize the lessons learned from the Fukushima Dai-ichi accident with the aim of further improving the safety and reliability of the fast reactor cycle. In 2012, scenario analyses were carried out on options for the nuclear fuel cycle policy according to various shares of nuclear energy supply in the medium term, mainly until 2030, in the long term plan making subcommittee of the Atomic Energy Commission [2]. In September 2012, at a meeting of the Energy and Environment Council comprised of the ministries concerned, it was proposed that the nuclear fuel cycle should be maintained and R&D on fast reactors, including the Monju prototype fast reactor, aimed at reducing waste volume and toxicity, and should be promoted although the basic policy is the realization of a society not dependent on nuclear power at the earliest possible opportunity [3]. Given the situation, this paper discusses the outline of the results of a long term scenario study, including the deployment of a fast reactor cycle in Japan, with the aim of clarifying the significance of long term fast reactor deployment.

2. TRENDS IN JAPANESE NUCLEAR POLICIES POST-FUKUSHIMA DAI-ICHI ACCIDENT

After the disaster, major changes in Japanese energy policies were made, such as the inauguration of the Energy and Environment Council in June 2011 by the ministries concerned, for the purpose of the development of global warming countermeasures and a new energy policy, and the establishment of the Fundamental Issues Subcommittee under the Advisory Committee for Natural Resources and Energy in October 2011, which was formed to develop a new basic energy plan under the Minister of Economy, Trade and Industry. Nuclear administrative agencies have also undergone review. The Nuclear Regulation Authority was established on 19 September 2012 transitioning from the Nuclear Safety Commission of Japan, and the Japan Atomic Energy Commission is under review for revision or abolishment.

Basic principles of the nuclear power use in the Framework for Nuclear Energy Policy [4], which was formulated in October 2005, was to have nuclear power continue to account for at least 30–40% of total electricity generation, even after 2030, and to aim at the commercialization of fast reactors by around 2050. The previous New Nuclear Policy Planning Council that compiled the basic principles conducted evaluations of the nuclear fuel cycle material balance and provided a comprehensive assessment from a multilateral perspective for four scenarios regarding the management of spent fuels: full reprocessing scenario (continuation of MOX fuel use in LWR scenario, transition to fast reactor scenario), partial reprocessing scenario, direct disposal scenario and intermediate storage scenario. As a result, it was decided to promote the nuclear fuel cycle that has the best potential for reduction of the environmental burden and for the effective use of uranium resources.

The Fundamental Issues Subcommittee, which was formed after the Fukushima Dai-ichi accident, estimated that the domestic electrical needs in 2030 will be approximately 1 trillion kW·h, which is 10% less than that of 2010, based on the premise of the following four directions: (i) fundamental reinforcement of energy and electricity conservation measures, (ii) accelerated development and use of renewable energies to the maximum degree possible, (iii) effective utilization of fossil fuels and (iv) reduced dependency on nuclear power wherever possible. The subcommittee engaged in discussion regarding dependency on nuclear power, considering options with nuclear power ratios of 15%, 20% and 25% by 2030, in addition to the option of zero nuclear power by 2020 or 2030 [5].

New Framework for Nuclear Energy Policy is also being discussed in the Energy and Environment Council. The Technical Subcommittee on Nuclear Power, Nuclear Fuel Cycle, etc., which was established in September 2011 by the Japan Atomic Energy Commission, conducted quantitative analyses and multilateral qualitative assessment of the aforementioned options laid down by the Fundamental Issues Subcommittee for the nuclear fuel cycle policy. As a result, it was reported that if the nuclear capacity is maintained or expanded in the medium to long term, the ‘full reprocessing scenario’ is the dominant alternative considering management and storage of spent fuels, disposal area for high level radioactive waste (HLW) and resource conservation, and if the future nuclear electricity capacity is uncertain, then the ‘partial reprocessing scenario’ is the leading candidate. On the other hand, if the policy is explicit about zero dependence on nuclear in the short term, then the full direct disposal scenario prevails. In addition, it describes that it is appropriate to proceed with the Rokkasho reprocessing project as planned unless the full direct disposal scenario is chosen and R&D of fast reactors should be continued if the full reprocessing scenario or partial reprocessing scenario is selected [2].

On 14 September 2012, the Energy and Environment Council composed of related ministries decided on a new energy strategy, Innovative Strategy for Energy and the Environment [3], and presented measures and goals to achieve realization of a society not dependent on nuclear power at the earliest possible opportunity, realization of a green energy revolution and ensuring a stable supply of energy. Upholding three guiding principles: to strictly apply the rules regarding the forty-year limitation of operation; to restart the operation of nuclear power plants once the Nuclear Regulation Authority gives safety assurance; and to not plan any new or additional construction of a nuclear power plant, this new strategy also describes that the Government will mobilize all possible policy resources to enable zero operation of nuclear power plants in the 2030s. Meanwhile it accepts maintaining the nuclear fuel cycle, including the Rokkasho reprocessing project and conducting R&D on Monju.

Subsequently, the Energy and Environment Council asked the ministries concerned to clarify specific goals or formulate plans for each item, such as the nuclear energy policy, the review on the Japan Atomic Energy Commission, Global Warming Countermeasures, and so on by the end of 2012. In the formulation of the new nuclear energy policy, an interim report that summarizes the R&D policy pertaining to the nuclear fuel cycle, such as the research plan for Monju and the study of direct disposal, was submitted to the Energy and Environment Council.

3. SIGNIFICANCE OF FAST REACTOR CYCLE DEVELOPMENT

Many countries still maintain their nuclear energy programmes after the Fukushima Dai-ichi accident and it is foreseeable that nuclear energy use worldwide will expand in the future. Figure 1 shows the prospects of global nuclear energy use and the cumulative global demand for natural uranium. Huge amounts of natural uranium, more than the total amount of conventional uranium resource, will be consumed by end of 21st century, according to low and middle level estimates of the JAEA [6] and other sources [7–11], and assuming that the LWR once-through option is continuously explored [12]. This result implies intensification of future competition for uranium resources, a remarkable rise in energy price and, at worst, depletion of the resource. In contrast, it is likely to constrain natural uranium consumption within the total amount of conventional uranium resource if fast reactors are deployed during the 2040s. The resource saving is a major inherent benefit of the fast reactor cycle. For those countries with scarce natural resources, such as Japan, fast reactor cycle deployment can

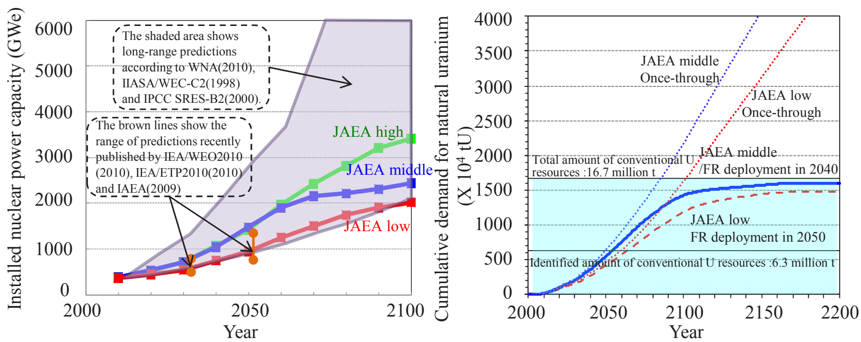


FIG. 1. Prospects of global nuclear capacity and cumulative demand for natural uranium.

release them from uranium imports and has a significant positive impact on their energy security.

Meanwhile, the environmental burden of reduction of radioactive wastes is also another inherent benefit of the fast reactor cycle [13]. The minimization of radioactive waste is a common and great challenge for countries relying on nuclear energy. Figure 2 shows a comparison of the amount of vitrified waste (i.e. HLW) between the LWR and fast reactor cycles. In the case of the fast reactor cycle with high heat efficiency, the amount of HLW would be decreased if compared to the LWR cycle case. With the application of high volume reduction technology in HLW production, such as fission product separation, the amount of vitrified waste will be significantly reduced. Additionally, the potential hazard from radioactivity of HLW is also considered important for public acceptance. As shown in Fig. 2, the recycling of not only U and Pu but minor actinides in the fast reactor has a possibility to reduce the potential hazard (radiotoxicity) drastically, approximately to 1/8 and 1/30, after about 1000 years of disposal, in comparison with current LWR vitrified waste and direct disposal of spent fuels, taking into account the achievement of a high actinide recovery rate at approximately 99.9%.

In addition to these inherent benefits of a fast reactor cycle, the economic effects are expected from the relatively stable supply and economical price of electricity. Figure 3 shows the electricity supply structure in Japan using an energy economic model analysis in the case of a continuous use of nuclear energy. The total electricity generation was kept to around 1000 TW·h/y, including about 15% of nuclear generation, up to around 2030. The transition from LWR to fast reactor was completed within 30 years in Japan. As for the economic effect, fast reactor deployment is effective to vitalize various industries, particularly metal

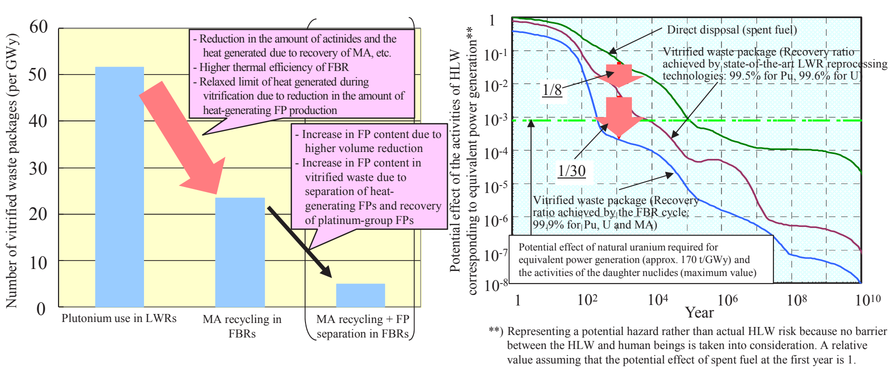


FIG. 2. Reduction effect of fast reactor cycle on the amount of HLW and the potential hazard of HLW.

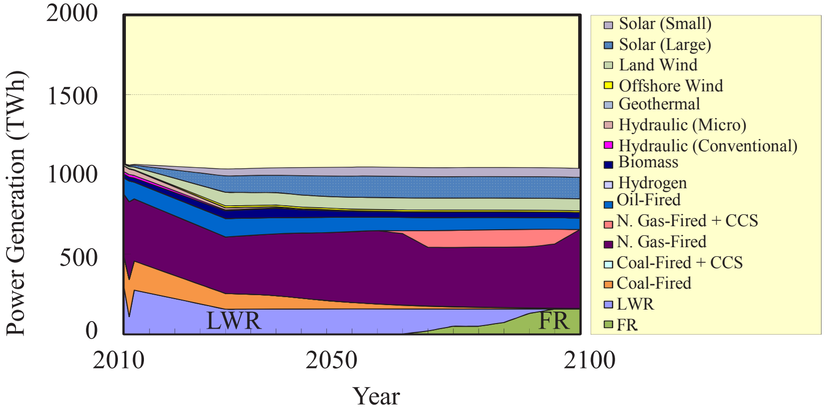


FIG. 3. Preliminary evaluation of the electricity supply structure in Japan.

industries, and accelerate their productivity. Total Japanese GDP improvement accumulation up to 2200 by fast reactor cycle deployment would reach several tens of billions of US dollars [14, 15].

As described above, the fast reactor cycle is an ideal application of nuclear fission energy, involving the efficient utilization of the resources, reduction of environmental burden and a positive economic effect.

4. JAPANESE LONG TERM FAST REACTOR CYCLE DEPLOYMENT SCENARIO

Evaluation of scenarios for long term nuclear energy use was conducted in accordance with assumptions made of several nuclear power generation capacities. This section introduces the evaluation example with a nuclear power generation ratio of 15% (corresponding to a nuclear capacity of 20 GW(e)) in 2030, which is one of the cases postulated following the Japanese energy policy debate after the Fukushima Dai-ichi accident. In the long term scenarios evaluation, two cases were studied in which either the nuclear capacity remains constant at 20 GW(e) after 2030 or it gradually reduces to zero after 2030, as shown in Fig. 4.

4.1. Major assumptions of long term scenario analyses

The major assumptions for the long term scenario evaluations are provided in Table 1. Those assumptions are based on the assumptions of short term

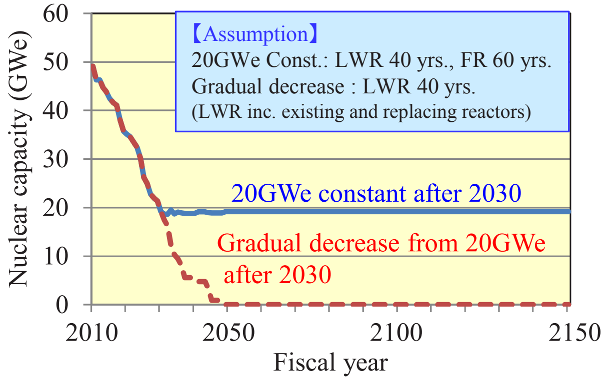


FIG. 4. Long term nuclear energy scenarios for 'Nuclear capacity 20 GW(e) in 2030'.

scenario evaluations until 2030, conducted by the Japanese cabinet office. The assumptions of facilities were applied adequately in each case as necessary (note that assumptions on the reprocessing plant were considered only in the recycling case) [16, 17]. The cross-cutting characteristic evaluation tool treating the whole nuclear fuel cycle supply chain developed in the Fast Reactor Cycle System Technology Development Project was used [18, 19].

4.2. Evaluation results for the '20 GW(e) constant after 2030' case

The nuclear capacities in the '20 GW(e) constant after 2030' case are shown in Fig. 5. In a full reprocessing scenario, Pu recycling in LWRs of maximum 18 GW(e) is being implemented over 40 years by using Pu recovered from overseas and from the Rokkasho plants before fast reactor deployment. Plutonium recovered from both LWR and fast reactor reprocessing plants is used and all LWRs are replaced by fast reactors in about 20 years after 2070, considering the decommissioning time of LWRs. On the other hand, in a full direct disposal scenario, Pu recycling in LWRs of maximum 16 GW(e) is being implemented over 10 years by using Pu recovered from overseas and from the Rokkasho plants.

TABLE 1. MAJOR ASSUMPTIONS

Facility	Item	Condition
FR	Introduction timing	Demonstration reactor: deployment in 2025 Commercial reactor: deployment in 2050 (to be deployed in accordance with Pu balance)
	Average burnup	Demo. reactor: 60 (initial)–150W·d/t, comm. reactor: ~150 GW·d/t
	Breeding ratio (Conversion ratio)	Demo. reactor: 1.1, comm. reactor: early phase: ~1.1, thereafter: 1.03 Burner reactor: ~0.6
	Capacity per unit	Demo. breeder reactor: 750 MW(e), comm. breeder reactor:1500 MW(e) burner reactor: 1500 MW(e)
	Lifetime/load factor	60y/~80%
LWR	Average burnup	After 2030: 60 GW·d/t
Fast reactor fuel fabrication facility	Commercial facility	Starts operation prior to fast reactor deployment 100t/y or 200t/y (constructed according to demand) Minor actininde upper limit 5%
Fast reactor fuel reprocessing facility	Commercial facility	Starts operation after fast reactor deployment, 100t/y or 200t/y (constructed according to demand), minor actinide recovery considered
	Spent fuel transportation after the cooling period	Transported to the reprocessing plant if reprocessing is available Continuously stored at the reactor site unless reprocessing is available
	Vitrification facility	Production conditions for vitrified waste: fission product oxide 10%, 2.3 kW/cask

TABLE 1. MAJOR ASSUMPTIONS (cont.)

Facility	Item	Condition
LWR reprocessing facility	Future reprocessing facilities after Rokkasho plant	Starts operation after the closure of Rokkasho plant Possible for reprocessing of MOX fuel and high burnup fuel: minor actinide recovery considered/mixes of all spent fuels before reprocessing
	Vitrification facility	Production conditions for vitrification: 1.25 cask/t
Storage facility	Spent fuel storage facility	Recycling: storage period of less than 40 years Direct disposal: storage period of 48 years, the capacity to be increased on demand
	HLW storage and management facility	Storage period: 50 years, constructed according to the near term plan, the capacity to be increased on demand
Geological repository	Vitrified waste	Starts operation in around 2037: upright position in hard rock
	Spent fuel direct disposal	Starts operation in around 2047: upright position in hard rock
Others	Out of core time	LWR cycle: 4 years or more/fast reactor cycle: 5 years or more
	Fuel cycle losses	Fuel fabrication: 0.1%, reprocessing: ~0.5% (LWR)/~0.8% (fast reactor)

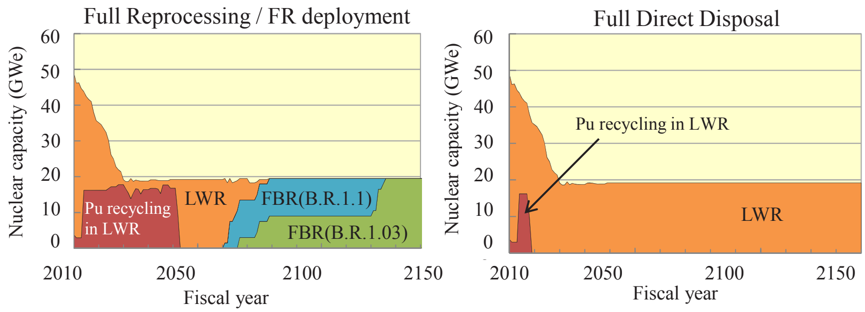


FIG. 5. Nuclear capacities of the '20 GW(e) constant' case.

The cumulative natural uranium demand is shown in Fig. 6. A full reprocessing strategy makes it possible to reduce natural uranium demand drastically after fast reactor deployment and to be fully independent of foreign natural uranium resources by around 2090, in addition to savings in natural uranium demand slightly before fast reactor deployment. The cumulative natural uranium demand is reduced to half that of full direct disposal by 2150. Figure 6 also provides details of the spent fuel stockpiles. Concerning the full reprocessing scenario, the spent fuel stockpile remains at 10 000–20 000 t until around 2060. Though the stockpile gradually decreases after 2060, the storage capacity may become tight, depending on the operational status of the reprocessing plants and it is necessary to raise the capacity until around 2080. Meanwhile, because the spent fuel stockpile increases up to 35 000 t before 2050 and becomes constant at about 17 000 t after 2080 in the full direct disposal scenario, additional storage capacity of 10 000–20 000 t will be required. If the Mutsu recyclable fuel storage centre and the pool of the Rokkasho plant are not available, up to 30 000 t of capacity will be required.

Radioactive waste volumes for geological disposal are shown in Fig. 7. Regarding the full reprocessing scenario, the low level radioactive waste (LLW) (geological disposal category) volume increases due to the deployment of reprocessing facilities, whereas the HLW (spent fuel and vitrified waste) volume decreases. As a result, the total volume of radioactive wastes for geological disposal reduces to less than half that of the full direct disposal scenario. In the full direct disposal scenario, spent fuels are disposed of in a geological repository and their amount continuously increases. Besides that, other categories of LLW are also reduced in a full reprocessing scenario because waste from operation of fast reactors for shallow land pit disposal and waste from decommissioning of fast reactors for shallow land trench disposal decrease, although waste for margin depth disposal increases due to the deployment of reprocessing facilities.

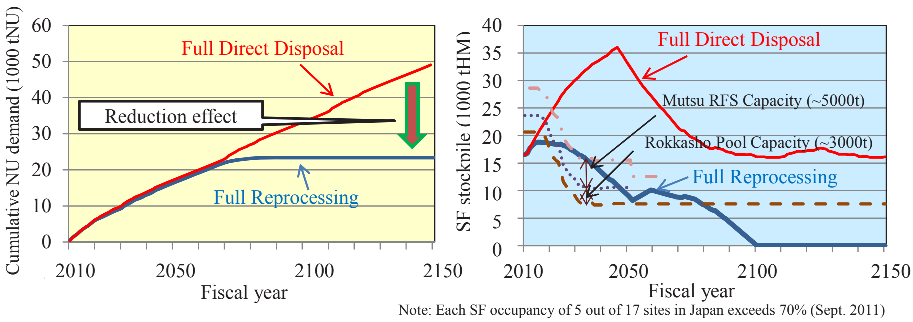


FIG. 6. Cumulative natural uranium demand and spent fuel stockpiles of '20 GW(e) constant' case.

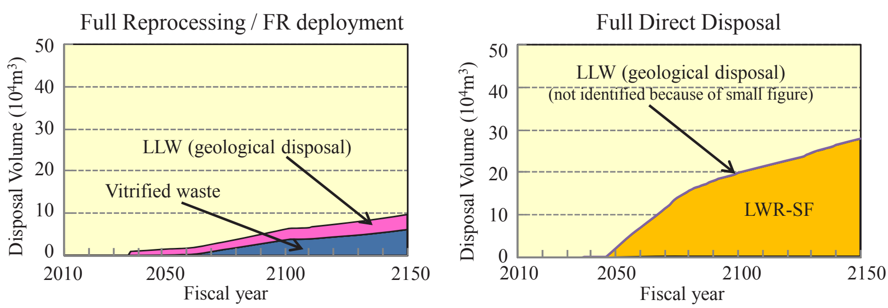


FIG. 7. Radioactive wastes for geological disposal of '20 GW(e) constant' case.

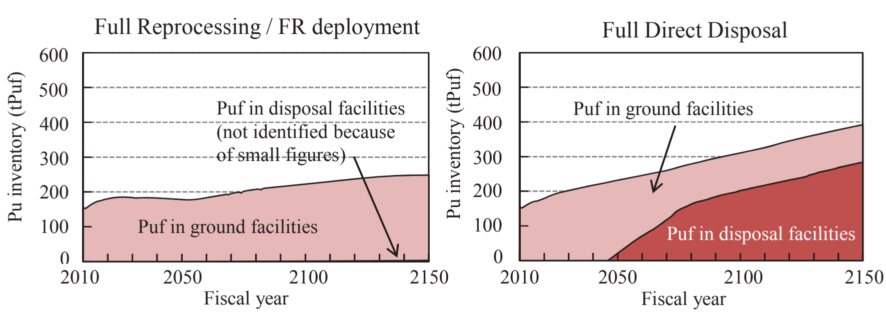


FIG. 8. Plutonium inventory in the '20 GW(e) constant' case.

The Pu inventory is a somewhat new topic for debate in Japanese nuclear energy policy. The evaluation results are shown in Fig. 8. A full reprocessing strategy enables lowering the Pu inventory in the whole fuel cycle by fast reactor deployment following the implementation of Pu recycling in LWRs. On the contrary, in the full direct disposal scenario, a large amount of Pu remains in spent fuels in the surface storage facilities undergoing cooling in the near term and in spent fuels disposed of in geological repositories in the long term.

4.3. Evaluation results for the 'Gradual decrease from 20 GW(e) after 2030' case (partial LWR reprocessing scenario and direct disposal scenario)

The nuclear capacities in the 'Gradual decrease from 20 GW(e) after 2030' case are shown in Fig. 9. In the partial LWR reprocessing scenario, Pu recycling in LWRs at a maximum of 18 GW(e) is being implemented for around 40 years using Pu recovered from overseas and from the Rokkasho plants. Meanwhile, in the full direct disposal scenario, Pu recycling in LWRs at a maximum of 16 GW(e) is being implemented over 10 years by using Pu recovered from overseas and from the Rokkasho plants.

Regarding the spent fuel stockpile in the partial reprocessing scenario, it remains at less than 20 000 t. Meanwhile, as the spent fuel stockpile will rise up to 35 000 t before 2050 in the full direct disposal scenario, additional storage capacity of 25 000 t will be needed, posing a bigger challenge when compared to the partial reprocessing scenario. If the Mutsu recyclable fuel storage centre and the pool of the Rokkasho plant are not available, up to 35 000 t of storage capacity will be required.

The radioactive waste volumes for geological disposal are shown in Fig. 10. In comparison with the full direct disposal scenario, the total volume of radioactive wastes for geological disposal in the partial reprocessing scenario reduces in spite of an increase in LLW (geological disposal category). As for the other categories of LLW, there is a no major difference in the total volume for disposal between the full direct disposal scenario and the partial reprocessing scenario.

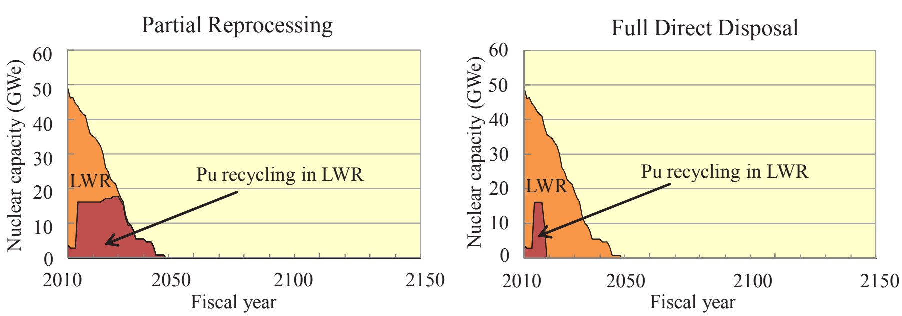
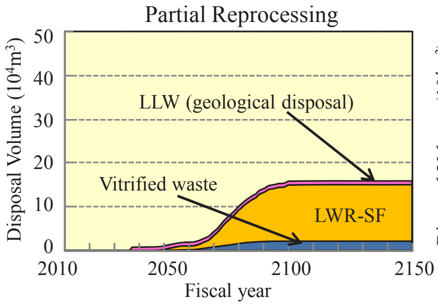


FIG. 9. Nuclear capacities of 'Gradual decrease from 20 GW(e)' case.



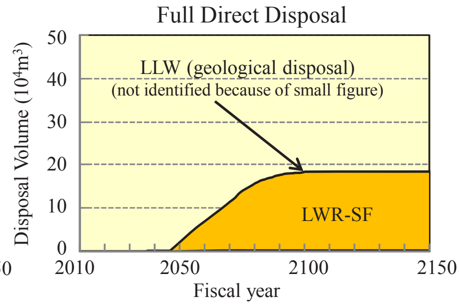
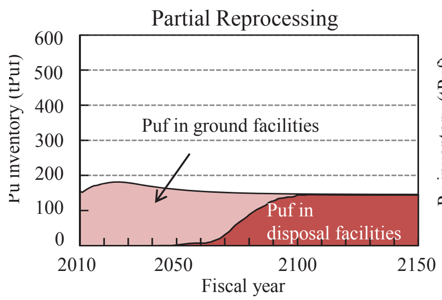


FIG. 10. Radioactive wastes for geological disposal of 'Gradual decrease from 20 GW(e)' case.



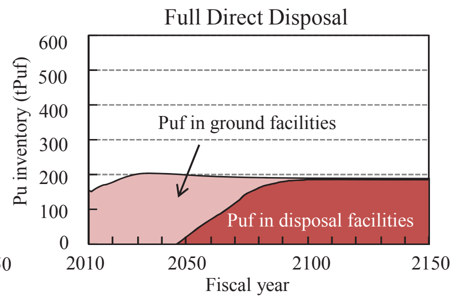


FIG. 11. Pu inventory in nuclear fuel cycle of 'Gradual decrease from 20 GW(e)' case.

The Pu inventory is shown in Fig. 11. The partial reprocessing strategy enables lowering the Pu inventory in the whole fuel cycle by the implementation of Pu recycling in LWRs. On the contrary, 200 t of Pu still remains in the spent fuels in surface facilities and/or disposal site in the full direct disposal scenario.

4.4. Evaluation results for the 'Gradual decrease from 20 GW(e) after 2030' full reprocessing (fast reactor deployment) scenario

The full reprocessing (fast reactor deployment) scenario in the 'Gradual decrease from 20 GW(e)' case is considered to pursue reduction of radioactive wastes, Pu inventory and radiotoxicity, which are left as the legacy of nuclear energy utilization. The main results are shown in Fig. 12. The operation of a fast reactor with 5 GW(e) (conversion ratio: ~0.6) after the implementation of Pu recycling in LWR with a maximum of 18 GW(e) enables lowering the Pu inventory in the whole fuel cycle to half that of full direct disposal (see Figs 11 and 12). Furthermore, reprocessing of all spent fuels lowers the total volume of HLW and geological LLW to about one half.

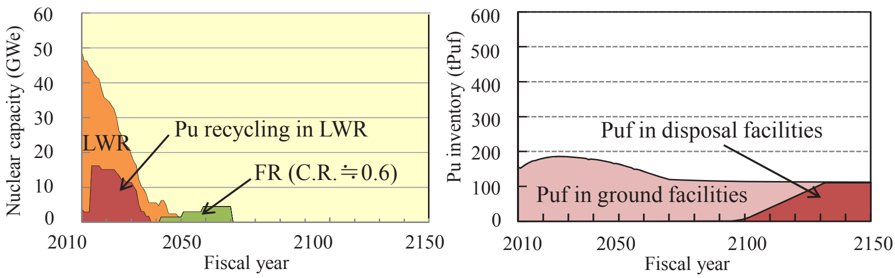


FIG. 12. Nuclear capacity and Pu inventory in the 'Gradual decrease from 20 GW(e)' case.

4.5. Summary of long term scenario studies

According to the long term scenario analysis results, the implementation of reprocessing is beneficial in reducing uranium demand, spent fuel storage, radioactive waste generation (especially for geological disposal) and Pu stockpiles from the Japanese nuclear fleet in both the '20 GW(e) constant after 2030' case and in the 'Gradual decrease from 20 GW(e) after 2030' case. If full reprocessing with fast breeder reactor/fast reactor deployment is realized, the benefits will become greater. Therefore, it is worth pursuing the current nuclear fuel cycle policy to fulfil the responsibility commensurate with a technologically advanced country. Moreover, the development of a fast reactor cycle for future deployment is favourable to realize a sustainable energy supply structure under the circumstances of a 'reduced reliance on nuclear power'.

5. CONCLUSION

Japan's future nuclear energy policy is still uncertain. This review of a scenario study of fast reactor deployment in Japan was conducted assuming various future states of nuclear power generation without being influenced by the current discussion on nuclear policies. The results revealed a need for implementation of reprocessing and development of fast reactors from the viewpoint of waste minimization, etc., in the medium to long term. Although Japan has been engaged in international cooperation on fast reactor development, our contribution to international cooperation and international standardization, focusing on the enhancement of safety and reduction of the radioactive waste burden, will be increasingly important in the aftermath of the Fukushima Dai-ichi accident. We intend to continue to build and propose fast reactor deployment scenarios and identify their characteristics.

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ECONOMICS OF FAST REACTORS: AN INDIAN PERSPECTIVE

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ABSTRACT

In the Indian energy scenario projections for the future, nuclear power through fast reactors is expected to play an important role, representing ~20% of the total installed electrical capacity by 2050. Successful operation of the Fast Breeder Test Reactor over 27 years, strong R&D executed in a multi-disciplinary domain and construction of a 500 MW(e) Prototype Fast Breeder Reactor (PFBR) based on an indigenous design have provided high confidence in the success of fast breeder technology. Beyond the PFBR, there are plans to construct six more FBRs, each of 500 MW(e) capacity. Towards this end, a systematic roadmap has been drawn up for improved economy and enhanced safety through a number of measures. The major features incorporated to achieve economy are twin unit concept, plant lifetime increased to 60 years in comparison to 40 years for the PFBR, reduced fuel cycle cost with higher burnup, number of steam generators reduced from eight to six, minimizing the use of 316LN stainless steel for the nuclear steam supply system, reduction in special steel specific weight requirements, compact plant layout, improved load factor, reduction in construction time by at least 2 years and co-location of fuel cycle facility. Beyond four reactors, a series of 1000 MW(e) capacity metallic fuel reactors with high breeding potential will be constructed and R&D activities have been systematically planned for metallic fuel development. The paper highlights in brief the results of studies carried out in some of the countries for cost reduction of fast breeder reactors. It also explains in detail the fast breeder reactor programme in India and highlights the measures taken to reduce the cost of future fast breeder reactors beyond the PFBR.

1. INTRODUCTION

Nuclear energy is an inevitable source of meeting the fast growing energy demands of India and its commitment to provide a better quality of life to all its citizens. For a large country, such as India, long term energy security, mainly based on indigenous resources is an important and inevitable need arising from economic, global environment and strategic considerations. These considerations will dictate the optimum composition of our energy mix. India's nuclear energy self-sufficiency extends from uranium exploration and mining, through fuel fabrication, heavy water production, reactor design and construction, to reprocessing and waste management. India has twenty thermal reactors

(18 PHWRs and 2 BWRs) in operation successfully with availability factors reaching up 90% consistently and seven more are under construction. India has a special interest in developing fast breeder reactors (FBRs) and the use of thorium as a source of energy as India has one of the largest reserves of thorium. India has a small fast reactor and is building a much larger one. It is also developing technology to utilize its abundant resources of thorium as a nuclear fuel. FBRs are more important to India than to other countries which have capabilities in nuclear power technologies. This is because of the nuclear resource profile in the country. India's current uranium reserves as regards the present state of exploration will only be able to support 12 GW(e) of generating capacity, which is not large. This is the main reason for developing fast reactors. With the same amount of uranium, which will support 12 GW(e) of generating capacity in PHWRs, and considering spent fuel reprocessing to retreive plutonium and residual uranium and use it in fast reactors, India will be able to feed an electric nuclear fleet capacity as large as 275 GW(e). This is due to the breeding potential of fast reactors, using the plutonium-uranium cycle. That shows the vital importance of FBRs under the Indian scenario, as compared to other countries.

Considering the existing reserves of coal in India and the rather meagre reserves of uranium, it remains a certainty that fast reactor based nuclear energy systems will have to be an important component of the Indian energy mix in the long term in order to meet the enhanced rate of energy consumption. Fast breeder technology is almost as old as nuclear power. To date, FBRs have been built and operated worldwide, ranging from research reactors having some tens of megawatts thermal output to the 1200 MW(e) Superphenix (SPX-1). The fast reactor technology has thus reached maturity with around 380 reactor-years of experience worldwide. Its commercialization vis–a-vis the established reactor systems such as PWRs and PHWRs will depend on its generating cost in fully developed form, with mature design and with the benefit of series production. For the success of fast reactors, efforts should be directed towards both reactor design and fuel cycle. The reactor has to combine safety with competitiveness. The fuel cycle, in particular reprocessing, has to be at an acceptable cost. Despite the indisputable need for FBRs in India, it is worth noting that reactors have to be economically attractive in the context of the present government approach for investments.

2. ECONOMIC COMPARISON (OTHER COUNTRIES)

2.1. Initial design reactors

Economic comparison data are available from France and the Russian Federation. For SPX-1, a prototype 1200 MW(e) unit in France, the construction cost/kW of installed capacity has been reported to be around 2.5 times that of a PWR operating in France [1] at that time. In the Russian Federation, the BN-600, a 600 MW(e) unit, lagged the WWER-1000 unit by a factor of 1.55 on specific capital costs. Considering the differences in power and site location, this difference decreases to a factor of 1.2–1.3 [2]. In brief, the first experimental prototypes of fast reactors did not match light water thermal reactors in capital cost or in unit energy cost.

2.2. Current reactors

It is unfair and misleading to make a comparison of prototype FBRs with matured PWR units. The PWRs have benefited from many years of experience in construction and, more importantly, from the benefits of scale arising from batch production. On the other hand SPX-1 and BN-600 were the first of a kind and were built as single units.

2.2.1. European Fast Reactor (EFR) versus SPX-1 and the economics of EFR

In Europe, the construction of fast reactors and design studies for larger units in France, Germany and the United Kingdom led to the development of combined European expertise on fast reactor technology. As a result of this collaborative venture, design of the European Fast Reactor (EFR) with a generating capacity of 1500 MW(e) was pursued. The major objectives set for the EFR were safety levels comparable with that of future LWRs, and potentially competitive electricity generation costs compared to future LWRs. Construction and operating experience of FBRs, in particular SPX-1, provided a wealth of information, allowing the simplification and optimization of the EFR. There has been considerable progress of knowledge in structural mechanics and design rules.

As a result, the design of EFR has achieved substantial investment cost reductions for the nuclear steam supply system, as illustrated in Fig. 1, which illustrates the specific weight of steel in t/MW(e) employed for the main system of SPX-1 and EFR [3]. Specific weight is recognized as a sound concept for economic comparison of designs. The design of the core and fuel for high burnup is an effective way of reducing fast reactor fuel cycle costs as this reduces the

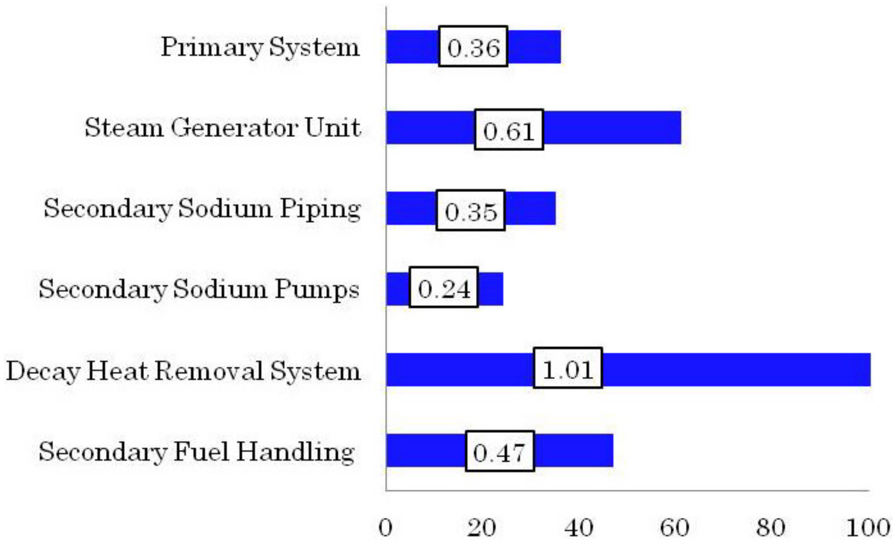

FIG. 1. EFR versus SPX-1: Comparison of specific steel weight in t/MW(e).

annual fuel requirement. The EFR core was contemplated to operate at target peak burnup of 20at.% as opposed to 8at.% for the first core of SPX-1. In the case of a PWR utilizing low enriched uranium, increasing burnup is accompanied by an additional cost arising from a higher enrichment requirement, thus partially off-setting the benefit from reduced annual fuel charges. The effect of burnup on the PWR and the FBR fuel cycle costs is shown in Fig. 2 [3].

In brief, the approach to economic competitiveness of EFR was based on the following:

(a) Investment cost savings due to:
- — Simplicity and compactness of the design;
- — Decreasing the number of safety graded systems while maintaining compliance with the required safety standards;
- — Reduction in number of components and systems, reduced component weights, and building sizes;
- — Improvement in construction, manufacturing and erection methods to allow shortening of the construction time.

(b) Fuel cycle cost savings arising from higher burnup.

The adoption of a 3 pumps/6 IHX arrangement for the primary system for the EFR against 4 pumps/8 IHX for the SPX-1 combined with an improved in-vessel fuel transfer system resulted in a remarkable reduction of the main

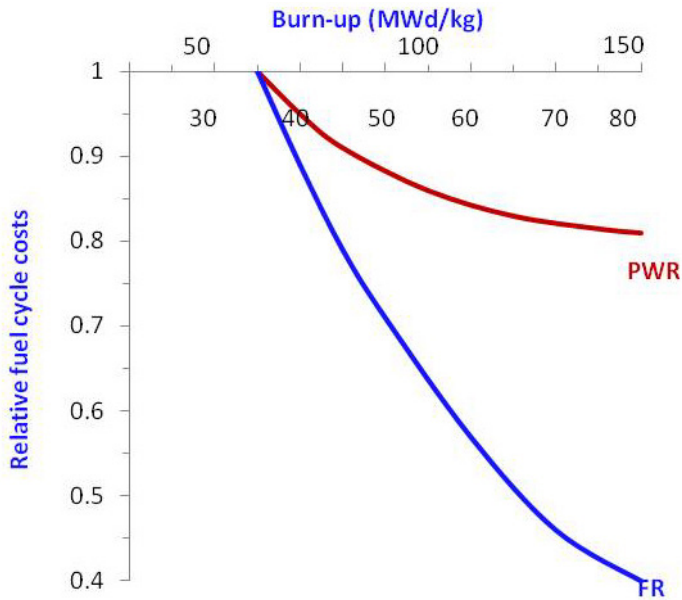

FIG. 2. Potential fuel cycle cost reduction by increasing average burnup in PWRs and fast reactors.

vessel diameter (17.2 m for the EFR compared to 21 m for the SPX-1) in spite of an increase in reactor thermal power by 20% as compared to the SPX-1.

The benefit of reduced component size and simplification in the secondary sodium circuit layout has also influenced civil works. The total volume of nuclear island buildings for the EFR is lower than that of the SPX-1.

The design exercise of the EFR has definitely improved the economics in comparison to the SPX-1, although it will still be a prototype in nature. As would normally be expected, any prototype is more expensive than a standard plant. There should be increasing benefits from series construction and learning effects.

The studies for series construction for the SPX-1 type indicate that investment cost could be as low as 58% of that of the prototype (Fig. 3) [4]. For the EFR, the savings corresponding to the elimination of its first-of-a-kind costs would be smaller that indicated by the SPX-1. However, savings corresponding to series effects would apply to the EFR. In addition, the effects of increasing the capacity of fuel fabrication and reprocessing facilities dedicated to the FBR will contribute to reducing fuel cycle cost. The comparison of generating costs between the EFR and PWRs is shown in Fig. 4 [3]. It shows that even when compared to the very efficient PWR (EPR), the EFR is very close to achieving competitiveness.

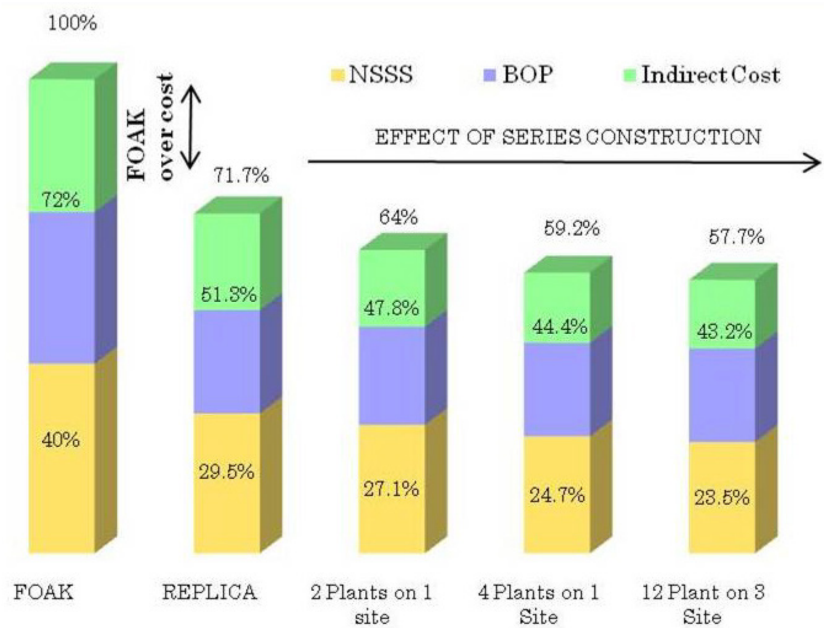

FIG. 3. FBR investment cost reduction.

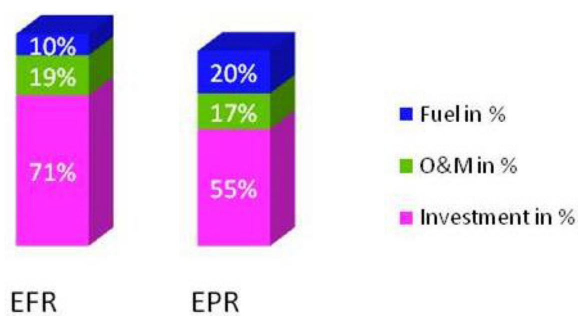

FIG. 4. EFR versus EPR generation cost comparison (costs normalized to 100% for EFR).

2.2.2. Russian fast reactors

In the Russian Federation, experience with the BN-600 has been used in designing the BN-800. The economic characteristics of the BN-800 design were improved in that the margins built into the BN-600 prototypes could be eliminated. The increase in power of the BN-800 reactor was accomplished with practically no increase in absolute material cost of the reactor [5]. The new

BN-600M reactor has significantly lower material consumption on account of the use of fewer heat transfer loops and an integral (tank) steam generator configuration (Table 1). The design of the new generation of thermal reactors with improved safety characteristics has reduced the difference in the economic performance of fast and thermal reactors.

TABLE 1. COMPARATIVE CHARACTERISTICS OF FAST REACTORS [5]

Characteristic	BN-600	BN-800	BN-600M
Thermal power (MW)	1470	2100	1520
Gross electrical output (MW(e))	613	793	647
Number of sodium loops	3	3	2
Material consumption in reactor (t/MW)	13	10.5	10.4 (8.23)[a]

[a] Figure in bracket corresponds to the design with an integral steam generator.
Note: Specific material consumption for WWER-1000 is 8.44 t/MW.

3. FBR PROGRAMME IN INDIA

FBRs are expected to play a major role in India's power programme and help in utilizing the country's large thorium reserves. Three decades of operation and maintenance of the Fast Breeder Test Reactor (FBTR) with no serious problems has provided confidence to pursue the FBR programme.

The Indian FBR programme was started with the establishment of a research centre (then known as the Reactor Research Centre) dedicated to the development of fast reactor science and technology and the decision to construct the FBTR, in collaboration with France, at Kalpakkam. The FBTR is a sodium cooled loop type 40 MW(th)/13.2 MW(e) experimental reactor and was commissioned in 1985. Today, the FBTR is one of the few fast neutron reactors operational in the world, and holds the promise of many more studies in this important domain. The experience gained in the construction, commissioning and operation of the FBTR, as well as worldwide FBR operational experience, and 30 years of focused R&D involving extensive testing and validation, material and manufacturing technology development and demonstration, peer reviews and synergy among the Department of Atomic Energy, R&D institutions and industries, have provided the necessary confidence to launch the Prototype Fast

Breeder Reactor (PFBR) of 500 MW(e) capacity. The PFBR is a pool type reactor where all the primary sodium components are in a single large vessel called the reactor assembly. The reactor has 2 primary sodium pumps, 4 intermediate heat exchangers, two secondary loops and four steam generators per loop. Austenitic stainless steel type 316LN/304LN is the major structural material for sodium components and piping and modified 9Cr-1Mo is the material for steam generators. The sodium temperatures are 820 K and 670 K for hot and cold pools, respectively. The plant design life is 40 years (load factor of 75%) with a potential to extend up to 60 years. The PFBR has many design features to achieve economy. A peak fuel burnup of 10at.% is targeted. A simple rectangular reactor building provides significant economy and construction advantages.

At present, the PFBR is in a very advanced stage of construction. The PFBR has been designed as a techno-economic demonstration of indigenous design and technology and is the forerunner of a series of fast reactors that are planned to be deployed. Co-location of the fuel cycle facility (fabrication, reprocessing and waste management) along with the reactor is also planned so as to minimize the fuel cycle cost of the PFBR and to exercise better control over fuel movement. This philosophy will therefore also be adopted in the planning of future FBRs at various sites. The dedicated Fast Reactor Fuel Cycle Facility for reprocessing of PFBR material is designed with additional capacity to handle the reprocessing needs of two more units of 500 MW(e) capacity each. Simultaneously with the construction of the reactor (PFBR), the fuel cycle has been addressed in a comprehensive manner and construction of a co-located fuel cycle facility has been initiated.

4. FROM THE PFBR TO FUTURE FBRS

Economic competitiveness is vital for the commercial deployment of fast reactors. Significant design efforts are necessary to reduce the capital cost of future FBRs coupled with enhanced safety. Therefore there is a challenge to identify the critical influential parameters that govern the overall cost and safety. Efforts are channelled into optimizing these with focused R&D, keeping in view with international experience.

The PFBR, being a prototype, is providing guidance for future design and construction. Enhanced safety and improved economics are twin objectives. The means to achieve economy has been identified as given below:

(1) Innovative design features to reduce specific weight of special steels;
(2) Multiple units at one site;
(3) Control of construction time;

(4) Raising the design temperature and improvement in cycle efficiency;
(5) Plant life extension from 40 years to 60 years;
(6) Load factor increased to 85%;
(7) Reduction in fuel cycle costs.

Lessons learned from the PFBR in plant layout, civil construction, manufacturing of nuclear steam supply system components in particular technical specifications, tender packages, regulatory review and component erection are being incorporated in the design and construction of future FBRs.

A detailed review of the capital cost breakdown of the PFBR indicates that reactor assembly, sodium circuits and fuel handling systems require closer scrutiny for possible reduction measures and there is little scope in the balance of plant due to level of standardization and maturity in its associated systems. Apart from the above, analysis of unit energy cost of the PFBR revealed further tangible benefits of enhancing plant thermal efficiency, fuel burnup and plant capacity factor, reducing construction time, multiple unit construction, policy measures on financial parameters such as depreciation rate, debt equity ratio, interest rate, etc. (Fig. 5). Through the above exercise and also based on experience with the FBTR, PFBR and FBRs worldwide and state of the art R&D with sodium cooled fast reactors, well defined optimization objectives/targets are defined for future FBRs.

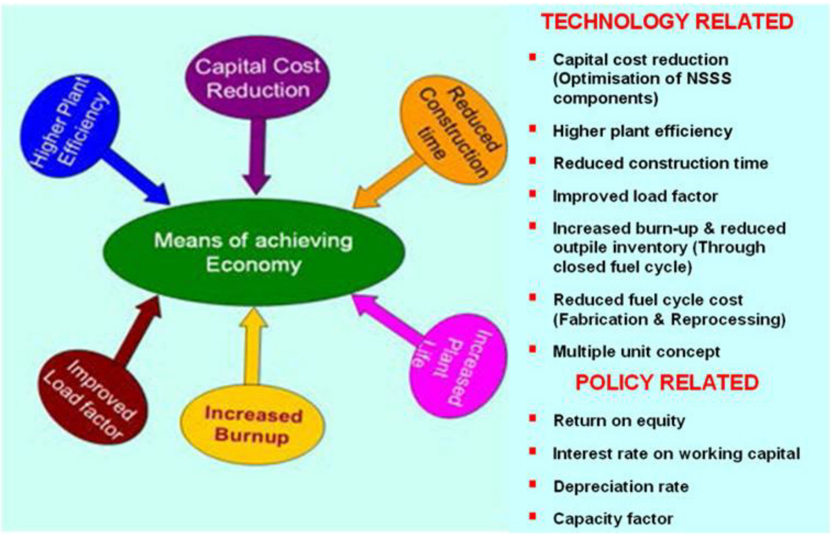

FIG. 5. Factors influencing overall cost.

Significant design changes are being contemplated in the design of major reactor assembly components with a view to optimizing the design and reducing capital cost. Further, the manufacturing experience with components for the PFBR has highlighted the focus areas that need simplification. For example, the adoption of a fully welded design concept for a grid plate (Fig. 6) with a reduced number of sleeves that support only the core subassemblies requiring coolant flow and a spike supporting arrangement for other peripheral subassemblies has offered considerable size and economic advantages, reflected in reduced overall manufacturing time as well. These changes have resulted in a 55% reduction in the overall weight of the grid plate. For reducing the shielding in the annular gaps between rotatable plugs and the roof slab in the top shield, thick plates with machined gaps offer potential. The other changes that are being actively pursued include options for a roof slab, change of configuration in the support of the reactor assembly, etc. Optimization of the permanent reactor assembly components has been completed and preliminary cost estimation indicates a saving of about 25% on specific weight, which in turn has strong linkage with the cost.

Similarly on the heat transport systems, from both economics and safety, the tube length in steam generators for future FBRs is planned to be 30 m in comparison to 23 m for the PFBR. Instead of design with eight steam generator modules with each tube of 23 m, a design with six steam generator modules with each tube length of 30 m is envisaged to reduce manufacturing time, enhance safety through about a 40% reduction in tube to tubesheet weld and about 25% reduction in cost. A module with fewer tubes is planned to be tested in the existing steam generator test facility.

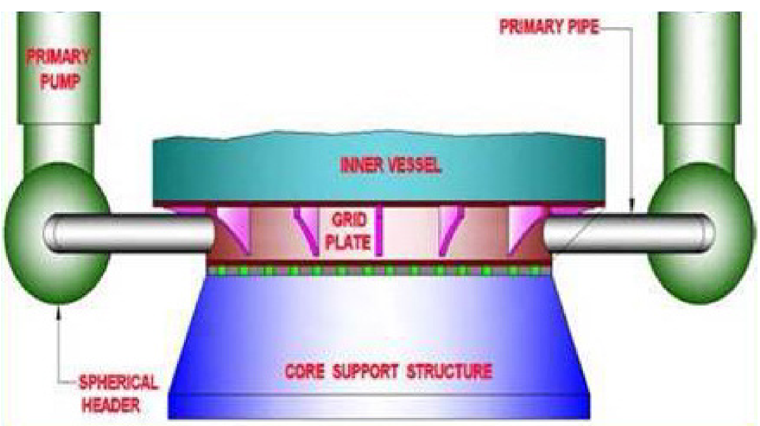

FIG. 6. Grid plate arrangements in future FBRs.

As part of cost optimization, the construction materials for the cold leg sodium and the hot leg sodium piping and components are proposed to be changed from 304LN and 316LN to 2.25Cr-1Mo and Modified 9Cr-1Mo, respectively. In view of this, development of sodium service valves in ferritic steel is being taken up with the participation of valve manufacturers in the country.

Future reactors will be on a twin unit concept as with Indian PHWRs. This will help reduce cost, as many buildings are shared between the twin units, resulting in improved economics. The cost studies indicate that a series construction of four at a given one or two sites would reduce the unit energy cost by about 35%. Moreover, the construction time could be brought down by two years. This results in a lowering of the interest during construction and hence a reduction in unit energy cost. The design life will be increased from 40 to 60 years, based on confidence in design analysis and development efforts. The increased design plant life has a favourable impact on lowering the depreciation component of unit energy cost.

Efforts are being directed towards bringing down the fuel cycle cost, which is linked to burnup (fuel as well as blanket). The target burnup for the fuel of the PFBR is 10at.%, but efforts are focused on increasing it to 15at.% and ultimately to 20at.%. Development of oxide dispersion strengthened steels for use as clad material for the fuel has been initiated to enhance the burnup to 20at.%. Co-location of the fast reactor fuel cycle facility gives substantial fuel cycle cost reduction, as shown in Table 2.

It has also been decided that only the PFBR and the next four FBRs will have mixed oxide as fuel. Subsequent FBRs will use metallic alloy as fuel. This decision is based on the potential of metallic alloy fuel to safely go for high breeding and high burnup. In line with this, the design of a 120 MW(e) experimental metal fuel reactor has been initiated.

TABLE 2. LEVELIZED FUEL CYCLE COST

Number of 500 MW(e) reactors (per fast reactor fuel cycle facility)	Levelized fuel cycle cost (%)
1	100
1+2	48
4	40
6	36

5. CONCLUSION

With good support from the Government and enthusiastic and committed involvement from academia and industry, the fast reactor programme in India is poised for huge improvement in the next few decades.

It should be reiterated that the PFBR is the first large size FBR being built in the country and therefore there is scope for cost reduction for future FBRs with the standardization of technology and series construction. Given that the FBR is an important component in India's quest for energy security and a link to the eventual utilization of thorium, there is merit in pursuing this technology.

ACKNOWLEDGEMENTS

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THE CONCEPT OF PHASED DEPLOYMENT AND CLOSURE OF THE NUCLEAR FUEL CYCLE ON THE BASIS OF FAST REACTORS UNDER CONDITIONS OF UNCERTAINTY WITH RESPECT TO THE FUTURE SCALE OF NUCLEAR POWER DEVELOPMENT

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Abstract

Usually the role of fast reactors (FRs) in nuclear power is associated with a special ‘mission’. In countries having a fast growing economy, the focus of the consideration, as a rule, is the ability of FRs to provide a high level of plutonium breeding. In countries having a stable economy, commissioning of FRs is usually associated with the ability to effectively burn or conduct multiple recycling of transuranium elements accumulated in the spent nuclear fuel from light water reactors. The practical implementation of the strategic missions requires considerable financial resources and time, and as a result, minimized interest from business and industry for FR systems. The authors analyse the idea of using already demonstrated sodium cooled FR reactors and mixed oxide fuel technologies in the Russian Federation, for example, to address the urgent problem related to the accumulation of WWER spent nuclear fuel. Clarification of the FR ‘mission’ in the Russian Federation and the development of innovative types of FR and closed nuclear fuel cycle technologies to fulfill this ‘mission’ is supposed to be done in parallel. The proposed phased approach minimizes financial and technological risks associated with the state support for the programme of FR system development and deployment, and in addition, may be of interest to the nuclear industry.

1. INTRODUCTION

Considerable success in fast reactor development has been achieved by the Russian Federation. The semi-industrial size BN-600 has been in successful operation at Beloyarsk NPP site since 1980. The construction of a new BN-800 FR is at the final stage at the same site.

After the Federal Target Programme “Nuclear Power Technologies of the New Generation for the Years 2010–2015 and for the Perspective to 2020” was approved in 2010 (NPTNG), a new phase for mastering FRs and closed nuclear fuel cycle (CNFC) technologies began in the Russian Federation. This programme for the period to 2020 includes the activities for constructing a mixed oxide (MOX) fuel manufacturing plant for the BN-800, for designing the BN-1200 commercial scale sodium cooled FR (SFR), as well as work for demonstrating FR technologies using heavy liquid metal coolants (lead–bismuth and lead). The scope of studies also includes R&D on mixed nitride uranium-plutonium fuel and elaboration of different options for the CNFC organization (both on-site and centralized variants).

It is supposed that at the end of programme implementation, there will be a possibility to choose the technology, and build an industrial module by 2025, which will consist of an NPP with two fast reactors of 1200 MW(e) power together with the SNF reprocessing and a fuel refabrication facility. It is also supposed that, starting from 2030, there would be the possibility of launching a large scale NPP development in this country on the basis of demonstrated technologies, up to the level of 100 GW(e) by the middle of this century.

Meanwhile, nuclear power in this country will be based on thermal reactor technologies (WWER and RBMK) and open nuclear fuel cycle. In accordance with the "Energy Strategy of Russia" approved in 2009, it is planned to increase the available power of NPPs (23.6 GW(e) as of 2012 to 52–62 GW by 2030).

Traditionally, solutions concerning the SNF from WWERs have been associated with the perspectives of mass construction of FRs, when plutonium from the WWER can be used for the initial loadings of FRs. Taking into account the NPTNG programme approved, the deployment and mass scale construction of innovative FRs is anticipated to be not earlier than 2030. With considerable uncertainty in predictions for the future, as well as the presence of natural risks for the implementation of innovative programmes, it may happen that solutions for the WWER SNF will be demonstrated only by the middle of this century. Meanwhile, the policy of delaying the solutions for thermal reactor SNF is characterized by a series of negative consequences, including:

— Decline in support for nuclear power as a whole by society because of a lack of decisions concerning the SNF and the accumulation of tonnes of plutonium in many countries in the world without a definite future, this being a sensitive material;
— Loss of energy potential of plutonium in the long term storage of SNF (owing to the decay of Pu-241 with a period of 14.5 years) and, as a consequence, accumulation of considerable quantities of radiotoxic Am-241, inducing serious radioecological problems in the future, when the option for reprocessing SNF from thermal reactors and recycling nuclear materials released in FRs would be available;
— The need to construct storage facilities for SNF from thermal reactors, the duration of storage being indefinite;

— The value of fuel components in the expenditures for NPPs is still not determined.

The authors suggest finding solutions for this emergent and system scale problem of SNF from thermal reactors in two approaches:

(a) First, to find ways for minimizing the risk of implementing the innovative FR and CNFC technologies in the framework of the NPTNG. This can be done, for example, on the basis of the analysis of results achieved and the system problems faced in the implementation of preceeding programmes for creating FR and CNFC technologies.
(b) Second, to open the floor for decisions on the fate of SNF from WWERs as soon as possible, for example, by using the FR and CNFC technologies that have been already demonstrated.

2. REVIEW OF PHASES OF FR AND CNFC TECHNOLOGY DEVELOPMENT IN THE RUSSIAN FEDERATION

In a history of more than 60 years in development of FR and CNFC technologies, four periods can be identified; these are sorted out by the choice of objectives, system requirements to the fuel parameters of FR, technological priorities and results achieved.

The period of achievements: 1950–1970s. This period was characterized by the high pace of national economic and power development. The Institute for Physics and Power Engineering, as the scientific supervisor of the fast reactor programme, determined the objective for development of FRs with sodium coolant (BN type) as the creation of a rapidly growing system of SFRs not limited in its progress by uranium resources. A requirement assigned to the fuel characteristics of SFRs was to achieve a system doubling time of ~6–8 years [1, 2]. The goal landmarks were also defined for the detailed parameters:

— Specific loading of fuel into reactor 2–3 t/GW(e);
— Plutonium breeding ratio 1.5;
— Detaining the fuel in the external fuel cycle for less than 1 year.

The research fulfilled during the same period by the Kurchatov Institute led to the alternative conclusion that for a guaranteed fuel supply for the fast developing nuclear power system, including thermal and FRs, it is necessary to have a system doubling of 3–4 years [3].

At that time, the requirement to achieve a very short doubling time for the FR was not officially supported. The priority task was set to demonstrate the operational capability of SFRs, with a high power density core ensuring achievement of a specific loading of fuel at the level of 2–3 t/GW(e), at a pilot facility. That task was accomplished successfully. The following SFRs were developed and commissioned:

— In 1973 — the first FR prototype (BN-350) with a thermal capacity of 1000 MW and 350 MW(e) used for gaining experience with the operation of a loop type design reactor and a high power density core with uranium oxide fuel.
— In 1980 — the BN-600 with a pool type design and a high power density core with uranium oxide fuel.

It should be noted that the wish to attain the priority goal of demonstrating the operational capability of FRs with sodium coolant and a high power density core in the shortest possible time led to a forced variant without MOX fuel in the BN-350 and BN-600 reactors because of lack of experience with this fuel in the country. The changeover to MOX fuel was planned as the next phase of the technology development.

In that period, the technology of water reprocessing of spent oxide uranium fuel from thermal reactors (WWER-440) and FRs (BN-350 and BN-600) was demonstrated at the semi-industrial scale RT-1 plant.

In general, the experience gained in construction and operation of both BN-350 and BN-600 reactors demonstrated the sodium based technology to be functional and the realization of high power density core with oxide fuel as a feasible possibility. The pool type layout of the reactor equipment was chosen as a reference for future reactors.

By the end of 1970s, the BN-800 FR project in a pool type layout was developed, based on MOX fuel with a high power density core and a moderate breeding ratio. This is considered as an intermediate step towards building a large nuclear power system with commercial FRs having high breeding ratios. The programme of complex demonstration of SFR and CNFC technologies at a commercial level was adopted, which included the construction of the RT-2 plant in the 1980s for water reprocessing of SNF from thermal reactors (WWER-1000), a plant for MOX fuel fabrication for FRs and four BN-800 units based on MOX fuels.

The 'diverse options' period: Late 1970s to 1986. In the late 1970s to the early 1980s, under conditions of decelerated economic development in the country, the Institute for Physics and Power Engineering was working on the justification of the following:

— Preservation of a phased approach towards the achievement of high standards of breeding; and
— Initiating the deployment of the industrial scale BN-1600 reactor with MOX fuels and a moderate breeding ratio, but with improved economic indicators, including a variant with a somewhat lower core power density [4].

In parallel, experts from Kurchatov Institute formulated an alternative objective and requirements for FRs, which were supported officially in 1984, namely, fuel supply for a system of thermal reactors, and, accordingly, achieving a high breeding ratio of 1.5 in the BN-1600. A possible way to attain this was considered to be a heterogenous core based on the uranium-plutonium oxide fuel with inserts from breeding metallic uranium [5].

The official priority became the development of the BN-1600 project with a breeding ratio of 1.5. The implementation of programmes for commercial deployment of MOX fuel production was delayed. However, the priority goal for that period was not achieved; the reasons were different, including the serious alterations of external conditions of NPP development that appeared after 1986.

The 'dull season' period: From 1986 to 2006. After the Chernobyl NPP accident in 1986, new requirements were developed in this country for NPPs with different types of reactor, including the BN type reactors. The radical economic reforms initiated at that time resulted in a decrease in demand for the construction of new electrical power plants nearly to zero, and this entailed a fall in interest in NPP development as a whole, and in particular FRs.

In the 1990s, the Institute for Physics and Power Engineering was working on resolving an urgent political task consisting of efficient utilization of excess plutonium withdrawn from the weapons progammes on the basis of the BN-800 reactor using MOX fuel. In this context of a new function, the fuel requirements for BN-800 were elaborated. In order to speed up the solution of this political task, it was suggested that the radial blanket be eliminated and, as a temporary variant, suspend SNF reprocessing [6].

At the same time, the Institute for Power Technologies advocated the need for large scale development of NPPs in the Russian Federation in the 21st century. To this end, a set of requirements in the areas of safety, non-proliferation, resource and nuclear wastes, the so-called 'natural safety' had been proposed. A new FR concept based on lead coolant with an on-site fuel cycle (BREST) meeting these natural safety standards was proposed [7].

The first order task in the area of FRs in this period were the modifications in the BN-800 project to meet the new safety standards, and a possible involvement thereof for achieving political arrangements. There was R&D work for the experimental justification of diverse technologies for MOX fuel fabrication. In

1997, a licence was obtained to resume the construction of BN-800; the first licence for the construction of an NPP following the Chernobyl accident.

Current period: 2006 to present. This phase began with a resolution from the Government to start funding new NPP construction in the Russian Federation, including the resumed construction of the BN-800 at the Beloyarskaya NPP. As early as 2000, the strategy for large scale NPP development for the period to 2050 was officially approved in the Russian Federation [8]. In accordance with this strategy, all the technologies developed for large scale nuclear power must meet the requirements of natural safety. As applicable to the fuel parameters of reactors, this suggests the following:

— Eliminate any reactor blanket (to reinforce protection against proliferation);
— Ensure a core breeding ratio of 1.05 (decreasing the reactivity margin in operation, which will prevent any prompt neutron runaway);
— Decrease the time needed for the external cycle to less than 1 year (to reduce the quantities of fissile materials in the cycle);
— Ensure multiple recycling of Pu in the reactor together with minor actinides accumulated in the NPP (to reduce the radioactive waste radiotoxicity to a level comparable with natural uranium radiotoxicity).

In 2010, the NPTNG programme was acccepted. In 2012, all work aimed at elaborating FRs and CNFC technologies that meet the requirements of natural safety were aggregated as a priority project “Proryv” (breakthrough).

This concept did not receive unanimous support in the scientific community. In 2011, an alternative view from the Kurchatov Institute on the pathways for NPP evolution in the Russian Federation in the 21st century was published, and the requirements for FR fuel characteristics were formulated [9]:

— Expedience in the use of blankets for a step-wise increase of BN-type reactor breeding ratios from 1.3 on MOX fuel to 1.5 on mixed fuels with metal inserts in the core;
— Delay the solution of the minor actinides problem to the future and provide dedicated molten salt reactors/burners for this purpose.

The scientific discussions around the objectives and tasks of FR development have not resulted in a revision of the NPP development strategy of the Russian Federation as accepted in 2000. It will need a long time before resolving a series of principles set in terms of strategy and any alternative concept. Meanwhile, there are several new challenges that can be solved in the framework of the possibilities provided by FR technology with sodium coolant and MOX fuel already demonstrated. One of these tasks is the implementation of political arrangements concerning the utilization of excess plutonium withdrawn from the weapons programmes. The BN-800 reactor will be included in the attainment of this goal. The commissioning of a production factory and a complete loading of BN-800 with MOX fuel are planned to be accomplished by 2017.

Since 2007, institutes have been working on a conceptual project for the BN-1200 reactor with a prospect for significant improvements in safety and economic indicators in comparison with the BN-800 [10]. It was supposed to attain the following in the area of fuel indicators:

— A breeding ratio of 1.2 with MOX fuels used during initial loadings;
— A low density core with an initial loading of 7.5 t of plutonium and an annual make-up of ~1.4 t.

In 2007, FTP “Nuclear Radiation Safety” was accepted. It is planned, under this programme, that before 2017, the Experimental Demonstration Centre will be designed and commissioned, with production capacity up to 300 t/year during the first phase of refinement of the upgraded technology for water reprocessing of SNF from the WWER-1000. The adapted and adjusted technology is assumed to be applied in the future on the commercial facility RT-2 plant, its commissioning being planned for 2025 as an orientation milestone.

3. SYSTEM SCALE PROBLEMS IN THE IMPLEMENTATION OF PREVIOUS PROGRAMMES FOR FR AND CNFC DEVELOPMENT

It can be seen from the analysis of the experience gained in the past that the scientific community’s views on the role of future FRs have changed, and that tasks for plutonium breeding of nuclear fuels formed 40–50 years ago as a priority have not been maintained, in fact, several more acute problems appeared. At the same time, the phased approach to attain the tasks of closing the fuel cycle with thermal and FRs, when resources are concentrated on critical pathways, as accepted in the period 1950–1970s, has given significant practical results. Actually, all what we have today, i.e. the successful operation of the BN-600 and the facilities under construction (the BN-800 reactor and the MOX fuel production facility), notably, are based on results of work performed during that period.

It was assumed that the industrial application of the MOX fuel technology would become the next major step to be implemented in the frame of phased approach. But this did not happen.

In our opinion, the main reason for that was lack of State funding of all work in the NPP domain after the Chernobyl accident, and the national economic reforms that lasted for about 20 years. In that period, there was no economic incentive for increasing the electrical capacity, including the nuclear one, and the future of nuclear energy following the Chernobyl accident looked gloomy.

However, another reason should be noted, associated with a negative evaluation by some experts in terms of the prospects for MOX fuels to achieve large scale power in the country.

We would like to call attention to the fact that delay in mastering MOX fuels in the early 1980s was caused to some extent by the requirement set at that time for a high breeding ratio of 1.5 for the commercial BN-1600 reactor. That entailed a need to cancel all alternatives to the MOX fuelled BN-1600 with a breeding ratio lower than 1.3.

Nowadays, in the framework of the official strategy of NPP development [8], the correspondence with requirements of natural safety suggests increasing the core breeding ratio to 1.05, which is equally difficult to achieve with MOX fuels. The increase of the breeding ratio to 1.5 in future SFRs is needed in the framework of the alternative strategy [9], which will also be unattainable with MOX fuels.

In this connection, questions arose on approaches used to define these contradictory requirements set for the breeding ratio and their justification.

The analysis carried out has shown that the authors of these contradictory requirements use similar methodological approaches. First, they deal with large scale nuclear power scenarios. Second, possible problems related to these scenarios are described and the role of FR in resolving these problems defined. Finally, proceeding from the prerequisites accepted, the requirements to be met by the FR fuel characteristics were set.

In the frame of such an approach, the justification of the requirements for the breeding ratio depends on the soundness of the authors' scenarios, and on the feasibility of the innovative technologies suggested by the authors in the future. This means the inevitable presence of risks.

The nature of risk is revealed through the long time requirement for the development of FR and CNFC technologies in the Russian Federation. For example, the high requirement for the doubling time of FRs was set in the 1960s–1970s based on an NPP scenario that never occured. In those years, the availability of raw materials was considered as the main problem for nuclear power. What happened, actually? In 1986–2006, there was no demand for new NPP development. Other important issues of modern NPPs were revealed as well and they concerned not only raw materials, or rather were not limited thereby; they were related to economic, safety, non-proliferation, accumulation of SNF, minor actinides and the main factor regarding nuclear energy, public opinion.

Now, in the framework of the new official concept of large scale NPP development, decisions on issues concerning current NPPs are delayed for the future, when mass construction of innovative fast reactors will take place. However, this means that there is risk, that the problems for current NPPs will remain unsolved in the next decades because of non-availability of innovative technologies on time, or because of the absence of the need for large scale deployment of NPPs at the time when innovative FRs would be mastered.

The situation is that, on one hand, the interest of the national nuclear industry to participate in the development of innovative technologies for the long term future has decreased, and on the other, additional measures for overcoming the real risks appearing at the new phase of FR and CNFC development in the Russian Federation are required.

The solution to both problems is viewed by the authors on the pathway to changing the paradigm in the methodology of defining the role of FRs and determining the requirements to their fuel characteristics and the choice of priority technologies to be deployed:

— From the **paradigm actually used**: The view on future nuclear power is formulated, with its problems described and requirements defined, then, work is carried out for creating innovative technologies for large scale nuclear power and in the process of implementation the problems of current NPPs are expected to be solved.
— To the **new paradigm**: The issues of current NPPs are identified, with requirements determined; first of all, those technologies are realized, allowing an accelerated resolution of current NPP problems and clearing the way for future NPP development. Simultaneously, the technologies used are modernized, and the new generation technologies are developed for the construction of large scale NPPs, correcting the requirements of these technologies in a permanent mode, as our concepts for the future are defined more precisely, and the views of the future and the real potential of the technologies used and those developed are shaped.

An example of the approach suggested is presented below.

4. ON THE FEASIBILITY OF RESOLVING THE SYSTEM PROBLEMS OF CURRENT NPPS BASED ON DEMONSTRATED TECHNOLOGIES FROM SFRs AND MOX

In worldwide practice, solutions of the problem of accumulated SNF are realized in two variants. First, there is the option of direct geological storage of

SNF adopted in Sweden and Finland. Second, there is the option of reprocessing the SNF, separating uranium and plutonium, with a single recycling in the form of MOX fuels in operational thermal reactors and with vitrification of high level waste for subsequent geological storage, as adopted in France. In the authors' opinion, neither of them would be acceptable for the Russian Federation.

The first option allows a complete solution of the problem of uranium SNF accumulation, but geological isolation of SNF together with plutonium precludes the possibility of any large scale development of NPPs in the Russian Federation after 2030.

The second option helps only partially in resolving the problem of SNF from thermal reactors. Indeed, in this option, the current annual volumes of the SNF from thermal reactors accumulated actually decrease 6–7 times, but in this case, a new problem arises. As a result of the recycling of plutonium in thermal reactors followed by long term storage of MOX SNF, the total mass of accumulated plutonium in the NPP is reduced by half, whereas the content of radiotoxic Am-241 therein is considerably increased. For countries where large scale development of NPPs with FRs is planned, such as China, India and the Russian Federation, inefficient burning of Pu with parallel accumulation of problems with Am is not acceptable.

In this work, the authors suggest a third option for resolving the problem of SNF from current NPPs based on the use of technologies that rely on the BN-type reactors and MOX fuels, already demonstrated in the Russian Federation. To a certain extent, the variant proposed is similar to that performed in France. The infrastructure for the option proposed also includes an industrial plant for chemical reprocessing of all uranium oxide SNF unloaded from WWERs and a facility for fabrication of MOX fuels from Pu separated in the reprocessing of SNF from WWERs. However, unlike the French option, the proposal is for fabrication of MOX fuels, not for recycling in existing WWER, but for a single-run in a small number of SFRs specially constructed for the utilization of Pu from SNF from WWERs (termed 'BN-utilizer').

As an orientation, the number of BN-utilizers is determined from the annual balance of Pu quantities accumulated in the WWER and the plutonium consumed for the fabrication of MOX fuels for the make-up of the BN–utilizer. For example, if we accept the level of Pu accumulation in a WWER of about 200 kg Pu/GW(e), and annual consumption of Pu for the manufacturing of MOX fuels for the BN-1200 of about 1200 kg Pu/GW(e), then this means that for the utilization of Pu received from six WWERs, it will be necessary to commission one FR having the same unit power level. Taking into account the additional demand for Pu for the initial loading of the BN reactor, the resultant relationship in this nuclear power system in terms of power between the WWERs and BN-utilizers may be defined in the range from 7 to 9. That is, if we have in the

NPP system about 10 per cent of power from BN-utilizers, a complete utilization of plutonium from the SNF originating from WWERs will become possible.

For example, to solve the SNF problem of all WWERs, it will be necessary to create the following fuel infrastructure, including:

— An RT-2 plant for chemical reprocessing of SNF from WWERs of about 800–1200 t/year capacity, depending on the total power of the WWER serviced from 40 GW(e) to 60 GW(e);
— An additional production line for the fabrication of MOX fuels from all separated plutonium of 8–12 t/year capacity; and
— Between 4 and 6 units of BN-1200-utilizers, instead of part of the WWER planned reactors, for the recycling of MOX fuels.

These BN-utilizers are supposed to be operated all the time with Pu produced in WWERs. That is, the infrastructure created will allow utilization of all WWER SNF accumulated in the NPP for energy generation, before the end of the WWER's lifetime.

As for the future MOX SNF from the BN-utilizers, it is proposed to determine it as dependent on market perspectives for the construction of new NPPs after 2030, and the preparedness of innovative technologies of FRs and CNFC for industrial deployment. With due account of uncertainties in our knowledge about the future, let us consider a wide range of possible scenarios of the NPP evolution, ranging from optimistic to pessimistic ones.

In the optimistic scenario of development, we assume that the market perspectives for the construction of new NPPs after 2030 will be fairly favourable. Innovative technologies developed in the framework of the NPTNG programme will be successfully demonstrated by 2030. When such conditions occur, the MOX SNF from BN-utilizers will be reprocessed in the frame of the infrastructure of CNFC created at that time, based on advanced technologies for large scale NPP development. The Pu separated from the SNF of the MOX BN-utilizers will be used for the fabrication of startup loadings of innovative FRs operating in the regime of the complete CNFC.

The second scenario also suggests that the market perspectives for NPP construction after 2030 will be favourable, but at the same time, for some reason, the industrial demonstration of innovative FR and CNFC technologies is postponed, for example, until 2040–2050. In this case, if the BN-1200 reactors with MOX fuels can be upgraded to a high enough level that they can meet the standards of safety and economic efficiency of the new NPP, then it is reasonable enough to assume that the MOX SNF accumulated from the BN-utilizers can be reprocessed, and that the Pu separated can be used for the fabrication of startup loadings of the new BN-1200 operating in the mode of the complete

CNFC. For the realization of a complete closed cycle for the new BN-1200, further development of the fuel infrastructure of the CNFC will be needed, namely, in addition to the production facilities required for reprocessing the SNF from the MOX of SFRs, as well as refabrication of MOX fuels from the separated plutonium.

Lastly, the third (pessimistic) scenario suggests that, owing to as yet unknown reasons, there would be no request for the construction of new NPPs after 2030. In this case, solutions to the problems related to the SNF from the MOX of BN-utilizers will have to be searched for in the frame of the existing NPP system, for example, via reprocessing of MOX SNF of BN-utilizers and organization of multiple recycling of plutonium in the form of MOX fuel, both in existing BN reactors and in the operating WWER reactors. It will be necessary to upgrade to a certain extent the fuel infrastructure created for the utilization of SNF from WWERs, with the addition of stages of reprocessing of MOX SNF, as well as refabrication of MOX fuels for the BN reactors and WWERs. The presence of a limited number of BNs in the NPP fleet will make it possible to organize the multiple recycling of plutonium in WWERs, owing to the use of low grade plutonium separated from the MOX SNF from WWERs, for the fabrication of MOX fuels for the BN reactors, ‘omnivorous’ in terms of the plutonium quality.

With the possible fate of SNF being determined for the BN-utilizers, a set of the system requirements to the fuel characteristics of first commercial FRs can now be defined.

Requirements for breeding ratio. The value of breeding ratio of the BN-utilizer in our case determines approximately a ratio of the quantity of Pu accumulated in the SNF from FRs to the quantity of Pu from the WWER used in the fabrication of MOX fuel for the BN. As a whole, any economically justified level of breeding ratio for the BN-utilizers with MOX fuels ranging from 1 to 1.3 will be fairly acceptable from the standpoint of implementing any of the three scenarios reviewed above for management of the SNF of the MOX BN-utilizers in the future.

Requirements for the fuel storage prior to reprocessing are determined separately for the SNF of WWERs and for the SNF of BN MOX.

In the case of the SNF of BN MOX, the requirements for storage periods will depend on the scenario for their utilization in the future. If reprocessing of SNF of SFR MOX is fulfilled in the frame of the future fuel infrastructure of the large scale NPP, then it is natural to suppose that the cooling time required will be estimated from the logic of minimizing the storage time to the level which ensures breeding of Pu for startup loadings of the new NPPs with FRs. Here, real requirements may be determined later, depending on the technological basis for the CNFC created at that time, and on the growth demand of NPPs with FRs in the period beyond 2030.

As for determining the requirements concerning the storage time of SNF from uranium WWERs, these requirements must be determined now. Two factors should be taken into account in this case. The longer SNF from WWERs is stored prior to reprocessing, the simpler and cheaper the reprocessing. However, on the other hand, the longer the storage time of SNF, the larger the quantity of radiotoxic Am-241 accumulating therein as a result of decay of Pu-241. The reprocessing of SNF from PWRs realized in France has shown that the optimal duration of storage before reprocessing of SNF from WWERs may be of 4–5 years, while the content of Am-241 in the SNF from WWERs is not large, and the Am separated in the reprocessing can be transformed into vitrified high level waste together with the fission products and the rest of minor actinides. In this case, Pu extracted in the reprocessing of uranium oxide SNF from WWERs will be relatively pure, and the production of MOX fuel for BN-utilizers may be organized without aggravating the radiation conditions.

Requirements for non-proliferation. Upon building the fuel infrastructure for the SNF utilization, it is deemed justified to take measures which make any access to sensitive nuclear materials very difficult. For example, in the reprocessing of SNF from WWERs, it is reasonable to separate Pu together with U in equal fractions. The MOX fabrication for BN-type reactors should be organized on a site in line with reprocessing of SNF from WWERs.

As for the non-proliferation issue in exporting nuclear technologies, it should be kept in mind that the basis for this export in the middle term will only be the WWER NPP. Construction of the fuel infrastructure for the utilization of Pu from the SNF from WWERs will allow exporting WWERs in a package with a proposal of a fresh fuel supply and the return of the SNF. Our standpoint is that this is the best option today for solving the problem of non-proliferation in the world of sensitive technologies and materials. The other variant is in the development of the concept suggested by the Russian Federation, i.e. creating international centres of the nuclear fuel cycle for SNF management.

Requirements for safety and economic aspects. It should be taken into account that the infrastructure proposed for the utilization of SNF from WWERs, including a limited number of BN-utilizers, must be regarded as a component of the existing NPP fleet. That is why there is no need to develop specific enhanced standards of safety, as in the case of technologies for large scale NPPs. From the economics standpoint, one should be aware that expenses for the construction and operation of the supposed fuel infrastructure, including the completed stage of radioactive waste management, should be part of the cost needed to solve the problem of SNF from current NPPs. It is important that the resultant cost of nuclear energy in this option should be acceptable for the consumer.

5. CONCLUSIONS

With approval in 2010 of the NPTNG programme, a new phase for mastering FR and CNFC technologies has started in the Russian Federation. The programme objective is to select and develop innovative FR and CNFC technologies for realization of a large scale NPP development starting from 2030.

Meanwhile, the authors put forward the idea of using demonstrated technologies in the area of BN-type reactors loaded with MOX fuels to solve the pressing problems related to the accumulation of SNF from WWERs. To this end, it is proposed to build new facilities for reprocessing the SNF from WWERs and for the fabrication of MOX fuels, as well as several power units of the BN-1200, for recycling MOX fuels with a capacity of up to 10% of the total NPP capacity.

The proposed option would allow a complete and efficient resolution of the problem of SNF from WWERs, with a minimum need for storage capacities and minimal quantities of radiotoxic Am-241 accumulated during storage of SNF from WWERs.

Moreover, the proposed option, in contrast to existing options (SNF disposal or MOX recycling in light water reactors), would preserve all plutonium accumulated in WWERs in a consolidated form (SNF MOX BN) as a startup resource for the deployment of large scale NPPs in any future scenario based on the use of advanced or innovative FR and CNFC technologies yet to be developed.

In the case of a pessimistic nuclear energy scenario, Pu from SNF MOX BN could be recycled in the form of MOX fuels in advanced WWERs (1/3 core) followed by multiple recycling in existing BN-utilizers.

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MAJOR FINDINGS OF THE IAEA'S INPRO COLLABORATIVE PROJECT ON GLOBAL ARCHITECTURES FOR INNOVATIVE NUCLEAR ENERGY SYSTEMS (GAINS)

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Abstract

The IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was established in 2000. INPRO cooperates with Member States to ensure that sustainable nuclear energy is available to help meet the energy needs of the 21st century. INPRO is part of the integrated services of the IAEA provided to those Member States considering initial development or expansion of nuclear energy programmes. This paper presents the major outputs of the INPRO collaborative project on Global Architecture of Innovative Nuclear Energy Systems with Thermal and Fast Reactors and a Closed Nuclear Fuel Cycle (GAINS (2008–2011)) which has developed a framework for the assessment of transition scenarios to future sustainable nuclear energy systems, validated it through application to several scenarios and highlighted the benefits and issues of collaboration among countries in making a transition to future sustainable nuclear energy systems. The paper ends with a brief outline of the ongoing follow-up INPRO collaborative project SYNERGIES (2012–2014).

1. INTRODUCTION

The IAEA fosters the peaceful use of nuclear power by supporting existing and new nuclear programmes around the world, catalysing innovation and building indigenous capability in energy planning, analysis, and nuclear information and knowledge. The IAEA provides integrated services to Member

States on nuclear power and the nuclear fuel cycle through the Planning and Economic Studies Section, the Integrated Nuclear Infrastructure Group and the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO).

INPRO was established in the 2000 as a flagship project of the IAEA, through a General Conference resolution, with the goal of ensuring a sustainable nuclear energy supply to help meet global energy needs in the 21st century [1]. INPRO's activities are focused on the concept of nuclear energy sustainability and support the development of long range nuclear energy strategies in Member States.

2. INPRO PROJECT 2: GLOBAL NUCLEAR ENERGY SCENARIOS

INPRO has developed a methodology for nuclear energy system assessment based on a comprehensive set of internationally agreed basic principles, requirements and criteria in the important areas of economics, safety, waste management, proliferation resistance, physical protection, environment and infrastructure [2]. Meeting the INPRO criteria in all of the areas ensures sustainability of the nuclear energy system and its high potential to meet growing energy demand throughout the present century.

Through nuclear energy system assessment, the INPRO Methodology helps Member States define their nuclear energy strategies for the near (up to 2030), medium (2030–2050) and long (2050–2100) terms [3]. Once the targeted nuclear energy system is assessed and adjusted to be sustainable, the question arises as to how to make a transition from the current fleet of reactors and fuel cycle facilities to such a targeted nuclear energy system. Assessment of such a transition — the process spread in time, essentially non-linear and typically involving cooperation with other countries — requires scenario analysis of national, regional and global nuclear energy systems in their dynamics, using material flow or, more generally, resource flow analysis codes.

INPRO project 2, Global Nuclear Energy Scenarios, has the objective of developing global and regional nuclear energy scenarios, on the basis of a scientific/technical analysis, that lead to a global vision on sustainable nuclear energy development during the 21st century. Specifically, this project helps participating Member States to identify and assess how to make a transition from the current fleet of reactors and nuclear fuel cycles to a future sustainable nuclear energy system and how national energy systems could contribute to, and benefit from, nuclear energy sustainability on regional and global levels. It highlights the role of collaboration among countries on the way to sustainable nuclear energy systems and aims to identify 'win–win' strategies for collaboration between suppliers and users.

This paper presents an overview and major findings of the IAEA/INPRO collaborative project Global Architecture of Innovative Nuclear Energy Systems with Thermal and Fast Reactors and a Closed Nuclear Fuel Cycle (GAINS), successfully completed under INPRO project 2 in 2008–2011 [4]. Section 3.8 highlights in brief the ongoing follow-up INPRO collaborative project on Synergistic Nuclear Energy Regional Group Interactions Evaluated for Sustainability (SYNERGIES (2012–2014)).

3. GAINS

The INPRO collaborative project GAINS was carried out in 2008–2011 by research teams from Belgium, Canada, China, Czech Republic, France, India, Italy, Japan, Republic of Korea, Russian Federation, Slovakia, Spain, Ukraine, the United States of America and the European Commission, with Argentina as an observer. The overall objective of GAINS was to develop a standard framework — including a methodological platform, assumptions and boundary conditions — for assessing transition scenarios to future nuclear energy systems regarding sustainability and to validate the results through sample analyses. The final report of the project has been prepared and approved for publication [4]. Major elements of the GAINS framework are summarized in brief below.

3.1. Synergistic heterogeneous world model

Previous studies of global nuclear energy scenarios used the so-called homogeneous world model wherein all countries in the world or a region were assumed to pursue the same policy regarding nuclear reactors and the nuclear fuel cycle and use the same facilities at a given time [5]. Different from that, GAINS has introduced an agreed upon model of the heterogeneous world comprising different nuclear strategy groups of countries (NGs) based on the spent nuclear fuel management strategy pursued for the back end of the nuclear fuel cycle, see Fig. 1.

For the purpose of GAINS analysis, three NGs were defined as follows. NG1 recycles spent nuclear fuel and pursues a fast reactor programme; NG2 directly disposes of spent fuel or sends it for reprocessing to NG1; and NG3 sends spent nuclear fuel to NG1 or NG2. The methodology applied in the analysis does not assign individual countries to groups, but allocates a fraction of future global nuclear energy generation to each group as a function of time to explore ‘what if’ scenarios. For the GAINS studies, the NG1:NG2:NG3 ratio was fixed at 40:40:20.

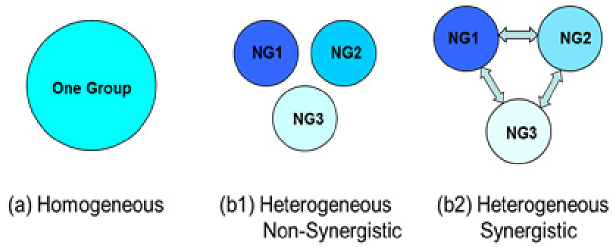

FIG. 1. Homogeneous and heterogeneous world models considered in GAINS [4].

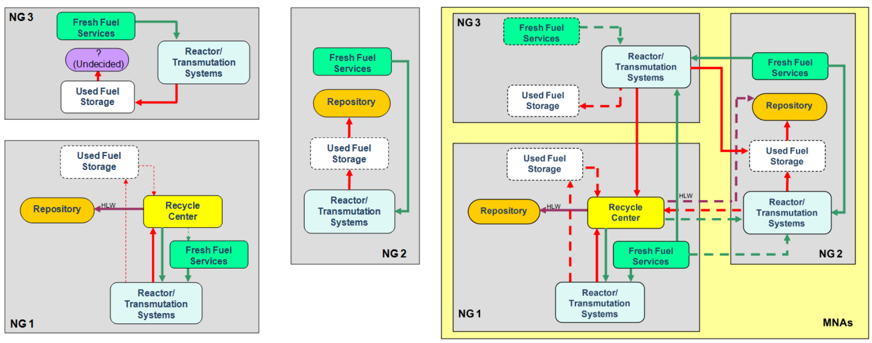

FIG. 2. Heterogeneous non-synergistic (left) and heterogeneous synergistic (right) storylines suggested in GAINS [4].

3.2. Three storylines for global nuclear energy system development

The GAINS framework provides for modelling of the three storylines for global nuclear energy system development: a convergent homogeneous world without any differences in nuclear energy system development strategies; a heterogeneous non-synergistic world based on self-reliance and preservation of local identities; and a heterogeneous synergistic world with rapid changes towards regional and global solutions (see Fig. 2). The synergistic model was a key model for the GAINS study.

3.3. Method for assessment of dynamic transition scenarios

A method for assessment of the dynamic transition of a nuclear energy system from its current state to a future potentially sustainable state is based on the application of 10 key indicators (KIs) coupled with related evaluation parameters (EPs) (see Table 1). Although bearing the same name, the KIs defined in GAINS are different from those defined in the INPRO Methodology for nuclear energy system assessment in that they are used to assess whether transition scenarios lead to sustainable nuclear energy systems rather than to assess sustainability of nuclear energy systems themselves. In most cases, the KIs and EPs defined in GAINS are integral over time or time dependent parameters designed to assess dynamic scenarios of nuclear energy system evolution. GAINS KIs build upon the INPRO Methodology but do not represent all of the assessment areas. For example, it is assumed that assessments of nuclear safety will be performed separately, apart from the GAINS analytical framework.

TABLE 1. KEY INDICATORS AND EVALUATION PARAMETERS OF THE GAINS FRAMEWORK [4]

No.	Key indicator (KI)/evaluation parameter (EP)	Units
	Power production	
KI-1	Nuclear power production capacity by reactor type	GW(e)
EP-1.1	(a) Commissioning and (b) decommissioning rates	GW(e)/a
	Nuclear material resources	
KI-2	Average net energy produced per unit mass of natural uranium	GW·a/ktHM
EP-2.1	Cumulative demand of natural nuclear material	ktHM
KI-3	Direct use material inventories per unit energy generated	kg/GW·a
	Discharged fuel	
KI-4	Discharged fuel inventories per unit energy generated	tHM/GW·a and m^3/GW·a

TABLE 1. KEY INDICATORS AND EVALUATION PARAMETERS OF THE GAINS FRAMEWORK [4] (cont.)

No.	Key indicator (KI)/evaluation parameter (EP)	Units
	Radioactive waste and minor actinides	
KI-5	Radioactive waste inventories per unit energy generation	m^3/GW·a (or kt/GW·a)
EP-5.1	Radiotoxicity and decay heat of waste, including discharged fuel destined for disposal	Sv/kW·h (or kW/t)
EP-5.2[1]	Minor actinide inventories per unit energy generated	kg/GW·a
	Fuel cycle services	
KI-6	(a) Uranium enrichment and (b) fuel reprocessing capacity, per unit of nuclear power production capacity	(SWU/a)/GW(e); and (tDM/a)/GW(e)
KI-7	Annual quantities of fuel and waste material transported between groups	ktHM/a
EP-7.1	Category of nuclear material transported between groups	Category (I, II, or III) [7]
	System safety	
KI-8	Annual collective risk per unit energy generation	Risk/MW·h; or qualitative discussion
	Costs and investment	
KI-9	Levelized unit of electricity cost (LUEC)	US$/MW·h
EP-9.1	Overnight cost for *n*th-of-a-kind reactor	billion US$; US$/kW(e)
KI-10	Estimated R&D investment in *n*th-of-a-kind deployment	billion US$
EP-10.1	Additional functions or benefits	Text providing qualitative description

[1] EP-5.2, KI-8, KI-9 and EP-9.1 were not used in GAINS; they will be applied in a follow-up study.

3.4. Internationally verified database of existing and prospective nuclear reactors and associated nuclear fuel cycles

The GAINS framework includes a database of existing and prospective nuclear reactors and related nuclear fuel cycles that extends the existing IAEA databases and takes into account the preferences of different countries. For reactors, the framework database includes:

- Low, medium and high burnup light water reactors;
- Heavy water reactors;
- Sodium cooled fast reactors with different conversion/breeding ratios (CR = 0.75; BR = 1.0, 1.2) and fuel burnups (from 31 GW·d/t up to 100 GW·d/t);
- Lead cooled fast reactor;
- Lead cooled accelerator driven system and molten salt reactor, both for minor actinide burning;
- ThO_2 and PuO_2 fuelled CANDU (heavy water) reactors; and
- ThO_2, ^{233}U and PuO_2 fuelled CANDU reactors.

For nuclear fuel cycles, the database includes:

- Once-through fuel cycle systems based on thermal reactors with different fuel burnups (business-as-usual scenarios);
- Combined system of a once-through fuel cycle with thermal reactors and a closed fuel cycle with fast reactors of different types (business-as-usual plus fast reactor scenarios);
- Combined system of a once-through fuel cycle and a closed fuel cycle with fast reactors and/or accelerator driven systems or molten salt reactors;
- Combined system of a once-through fuel cycle and a closed (U-Pu)/Th fuel cycle with fast reactors.

For each of the above mentioned options, fuel cycle conditions are specified as appropriate, including:

- Reactor types and their fuel burnup;
- Plant lifetime and load factors;
- Proportions of each reactor type and their changes with time;
- Process time for front end and back end fuel cycle stages;
- Tails assay of uranium enrichment;

- Cooling time for spent nuclear fuel in nuclear power plant storage;
- Out of reactor period for discharged nuclear fuel;
- Capacity of reprocessing facilities; and
- Heavy metal loss in reprocessing.

3.5. Selected long term nuclear energy demand scenarios

The GAINS framework included two long term nuclear energy demand scenarios based on the high and low estimations of nuclear power deployment until 2030 by the International Panel for Climate Change and the IAEA and expected trends until 2050 based on forecasts of competent energy organizations (see Fig. 3). These scenarios can serve as reference points in analyses of the global nuclear system.

3.6. Documented framework base cases

The GAINS framework described above was used to calculate global nuclear energy scenarios, each corresponding to moderate and high demand growth rates, for:

- A business-as-usual plus fast reactor scenario in a homogeneous world model;
- Business-as-usual plus fast reactor scenario in heterogeneous non-synergistic and heterogeneous synergistic world models[2].

The resulting eight scenarios are referred to as framework base cases in Ref. [4]. Sensitivity analysis to modelling assumptions, including tails assay, fuel burnup, cooling time, changes of fast reactor type and changes of reprocessing capacity deployment mode were carried out for the framework base cases. All scenarios were analysed and compared using the key indicators and evaluation parameters shown in Table 1.

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[2] In the synergistic case addressed in GAINS, NG3 receives fresh fuel from NG2 and NG1 and returns the associated spent fuel to those groups. As an extension of the GAINS study, one could consider NG2 and NG3 to both ship their spent nuclear fuel to NG1 for reprocessing and recycle in fast reactors in NG1. Such 'asymptotic' cases can be effectively addressed by the homogeneous world model.

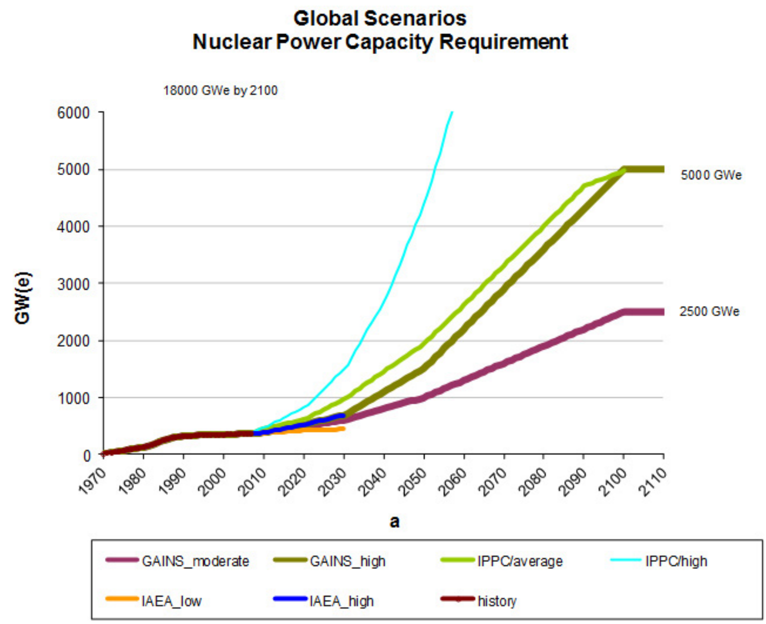

FIG. 3. GAINS scenarios for nuclear energy demand growth [4].

In addition to this, other innovative nuclear energy system scenarios were analysed in the homogeneous world model, including:

- Scenarios with fast reactors or heavy water reactors using thorium fuel cycle to reduce natural uranium requirements;
- Scenarios with accelerator driven systems or molten salt reactor to reduce minor actinide inventory, and others.

The output data and assumptions for all scenario calculations performed in the GAINS project are documented as Excel files, named the scenario templates, which are provided on a CD accompanying the project final report [4]. Of the total, 55 scenarios with nuclear reactors and fuel cycles of different types are included (see Annex 3 in Ref. [4]). These data allow analysts to reproduce each of the preformed scenario calculations using different codes and to build on them in further studies of transitioning to globally sustainable nuclear energy systems.

3.7. Major findings

A major finding of the GAINS collaborative project is that the world is likely to follow a heterogeneous model, at least within the 21st century.

Although fast reactors in a closed nuclear fuel cycle are known to provide sound solutions for minimization of the spent fuel inventory and expansion of the nuclear material resource base, at least within the 21st century, the world is most likely to follow the heterogeneous model, wherein only a few countries will be able to develop and commercialize fast reactor based systems. The reason is that for small (less than 30 GW(e)) programmes of fast reactor/closed nuclear fuel cycle deployment, the economic benefits from their introduction would be substantially lower than the amount of investment needed for their development, design, demonstration, licensing and deployment.

Indeed, Ref. [8] indicates that the worldwide investments already made in the development and demonstration of the sodium cooled fast reactor technology exceed US $50 billion. The cost of the former USSR research, design and demonstration (RD&D) programme on development of sodium cooled fast reactors is estimated by Russian experts at US $12 billion. Yet, the technology will require additional significant efforts and funding for reaching the commercialization stage.

The results of calculations of a payback time for RD&D investments needed to develop and implement fast reactors and associated closed nuclear fuel cycles are shown in Fig. 4, versus the market size for such reactors. In this calculation, it was assumed that RD&D investments for developing fast reactors and the associated fuel cycle technologies would range between US $10 billion and $40 billion, and that fast reactors can be built at US $2000/kW(e).

From Fig. 4 it is clear that payback time for investments appears reasonable only when the targeted capacity of fast reactors for deployment is large and the required investment needed to bring fast reactors to deployment is reasonably low. For example, if the new capacity based on fast reactors with a closed nuclear fuel cycle is 30 GW(e) or more, the required investments can be recovered within

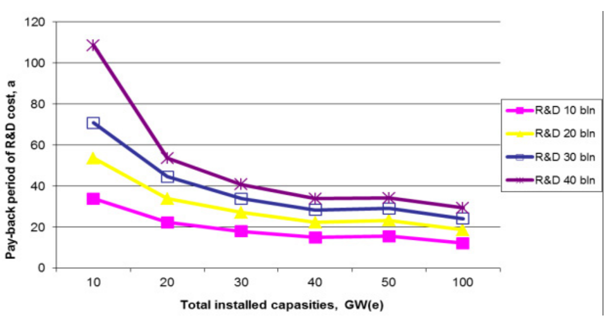

FIG. 4. Payback period for RD&D investments in fast reactors and associated closed nuclear fuel cycles for different installed capacities [4].

20 years if they are about US $10 billion and within 40 years if they are about US $40 billion. Conversely, if the market size is only around 10 GW(e) then the RD&D expenditures of US $40 billion are harder to justify as they will not be recovered within the 21st century.

The conclusion from this study is that for small programmes of fast reactors/closed nuclear fuel cycle deployment, the economic benefits from their introduction would be substantially lower than the amount of investment needed for their development, design, demonstration, licensing and deployment. Only a few countries in the world with targeted large nuclear energy programmes (30 GW(e) for fast reactors only) can bear the burden of technology development of fast reactors/closed nuclear fuel cycle. Therefore, the global nuclear energy system would follow a heterogeneous world model, at least, within the present century.

Other findings of GAINS are as follows:

(1) Although only a few countries will master the innovative technologies of fast reactors and the closed nuclear fuel cycle within this century, all others could benefit from this if they follow a synergistic approach, i.e. they send their spent nuclear fuel for reprocessing and recycle in fast reactor programmes implemented by ‘fast reactor countries’. In this, progressive accumulation of spent nuclear fuel on a global or regional scale could be mitigated or even reversed to limit the inventory of such fuel to minor actinides and fission products or only fission products, if minor actinides are further incinerated in dedicated transmutation systems.
(2) The above mentioned synergistic approach within a heterogeneous world offers potential benefits associated with reducing both the inventories of direct use material (plutonium) and the number of sites using sensitive technology for fuel reprocessing. GAINS calculations have shown that, under a synergistic approach, the global plutonium inventory could be reduced down to a minimum stock needed for nuclear energy system operation (see Fig. 5).
(3) A synergistic approach could also secure natural uranium savings of up to 40%, compared to the heterogeneous non-synergistic case.
(4) Countries that do not pursue fast reactor programmes could benefit from the synergistic approach as it results in reduced requirements to long term spent fuel storage and ultimate disposal of waste. Even if fission products are returned, their volumes will be substantially smaller compared to spent fuel before reprocessing and, additionally, proliferation concerns will not exist for storage or final disposal of such waste.

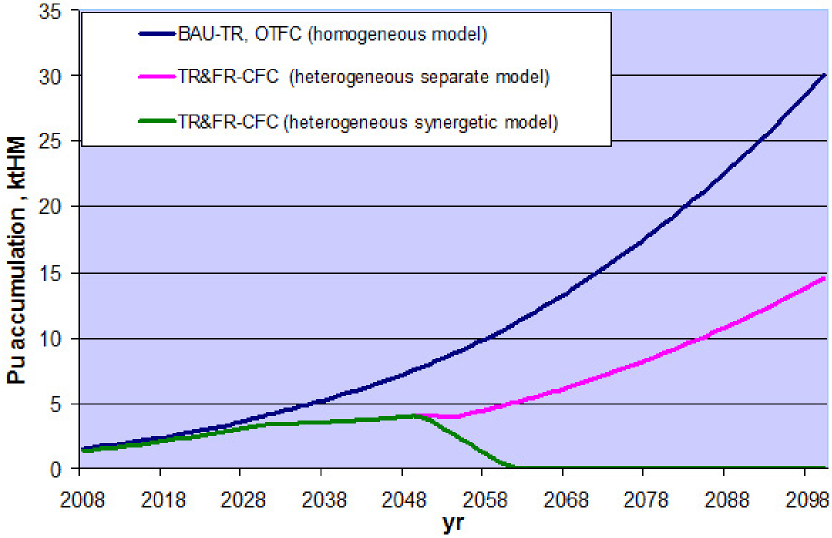

FIG. 5. Excess plutonium inventory in the nuclear energy system (surplus to the minimal plutonium stock needed for system operation) for GAINS 2500 GW(e) scenario from Fig. 3 (BAU-TR: business as usual with thermal reactors, OTFC: once-through fuel cycle, TR&FR CFC: closed nuclear fuel cycle with thermal and fast reactors) [4].

(5) Reprocessing capacity requirements will increase for the 'fast reactor countries' in the case of spent fuel shipment from other countries. However, in this case they would acquire larger fissile resource to continue with expansion of their closed fuel cycle and fast reactor programmes, benefitting from smaller RD&D risks and shorter payback time on investments.

(6) Within the considered synergistic approach, all countries could benefit from the lower cost of fuel cycle services owing to economies of scale and/or economies of accelerated learning. As natural uranium resources are also being saved through synergistic cooperation, all countries could also benefit from longer lasting lower costs of natural uranium.

(7) The reprocessing capacity requirements will increase for NG1 countries in the case of spent fuel shipment from non-NG1 countries. However, in this case, NG1 countries would acquire a larger fissile resource to continue with expansion of their closed fuel cycle and fast reactor programmes, benefitting from smaller RD&D risks and shorter payback time on investments (see Fig. 4).

3.8. Follow-up activity

While GAINS made some generic conclusions on economic and other benefits and possible risks of synergistic collaboration among countries, these conclusions are of a preliminary nature. Further in-depth studies will be necessary to understand the driving forces behind such collaboration and possible impediments in a collaborative way to further globally sustainable nuclear energy systems.

A follow-up collaborative project Synergistic Nuclear Energy Regional Group Interactions Evaluated for Sustainability (SYNERGIES), started by IAEA/INPRO in 2012, will focus on in-depth evaluation of synergistic collaborative scenarios of nuclear fuel cycle infrastructure development. The objectives of SYNERGIES are to quantify the benefits and issues of collaboration among countries in transitioning to globally sustainable nuclear energy systems and identify those transition scenarios which offer a ‘win–win’ strategy for both technology holders and users [9]. To achieve this goal, SYNERGIES will use the GAINS framework and modify or amend it as appropriate. The specific objective of SYNERGIES is to identify short term (2012–2030) and medium term (2030–2050) collaborative actions capable of developing pathways to long term sustainability.

4. CONCLUSION

This paper presents an overview and major findings of INPRO’s collaborative project GAINS, which has been implemented under the INPRO project Global Nuclear Energy Scenarios in 2008–2011.

GAINS has developed a framework for the assessment of transition scenarios to future sustainable nuclear energy systems, validated it through application to several scenarios based on thermal and fast reactors and a closed nuclear fuel cycle, highlighted benefits and issues of collaboration among countries in making a transition to future sustainable nuclear energy systems, and thoroughly documented all models, assumptions, data and results. The final report of the project will be published in the IAEA Nuclear Energy Series.

The major findings of GAINS indicate that collaboration among countries in the fuel cycle back end is crucial to moving towards global sustainability of nuclear energy systems and suggest multilateral, perhaps regional, approaches be considered further in this area. The GAINS studies will be furthered in the ongoing INPRO collaborative project SYNERGIES, which will identify and evaluate mutually beneficial collaborative architectures and the driving forces for, and impediments to, achieving globally sustainable nuclear energy systems.

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COMMERCIAL US VENDORS FOCUS ON REDUCING THE COST OF FAST REACTORS

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Abstract

Fast reactor development was originally motivated by the perceived scarcity of uranium and fast reactors were designed to be integrated with fuel cycle reprocessing. Although these are still important considerations, several commercial companies in the United States of America are exploiting other key characteristics of the fast neutron spectrum to design reactors that offer significant capital and energy production cost reductions. Corporate operating philosophies, funding mechanisms, target markets, reactor fuels, coolants and designs from each of these companies are vastly different. Despite this, the companies are each focusing on one or more of the following fast neutron spectrum characteristics: compact designs, inherent safety characteristics, improved efficiencies due to high temperatures, extended core lifetimes and/or high fuel burnup. The challenge they perceive is to design reactors that customers will purchase primarily because they are the most cost effective for their energy needs.

1. INTRODUCTION

This paper considers the business models of four US based companies that are invested in fast spectrum reactors. Although each company is different in various ways (e.g. reactor designs, markets and funding mechanisms), they have settled on fast reactor designs because of the benefits they offer for the markets they envisage. All of these companies are struggling in a political climate that has seen an ongoing decline in US public and Government support for fast reactors since 1974 [1], and especially since the shutdown of the Integrated Fast Reactor programme in 1994 [2], which makes these companies' resolve to pursue fast reactors all the more interesting. Without a current political path towards allowing reprocessing capabilities, it also means that the motivation for fast reactor designs (at least for these companies) was not driven by the reprocessing direction that originally guided the US fast reactor programme. The fast reactor designs in this paper contain no uranium blankets and do not rely on fuel reprocessing for their marketed benefits. The primary business benefits accrue from the fast reactor design characteristics themselves.

These fast reactors operate using larger energy densities that often enable more compact designs and readily lend themselves to modularization. Output temperatures are typically higher than for existing or advanced light water reactors (LWRs), thereby enabling power generation equipment to run at higher operating thermal efficiencies. The higher temperatures also lend these power sources to process heat applications such as petrochemical refining and water desalination. The neutron balance requires a higher ratio of fission material enrichment to start, but these reactors burn actinides that are generated as part of the fuel burning process.

Any of these reactors are, of course, capable of burning reprocessed fuel; and some of these companies have an eye towards reprocessing methods that do not separate plutonium — a perceived reason why US policy has been directed against reprocessing efforts. If any of these methods should one day find favour and can be demonstrated to be economically viable, fast reactors have additional benefits that can accrue when spent fuel, either from LWRs or from the fast reactors themselves, can be reprocessed and burned as new fuel in these fast reactors.

The following sections present high level summaries of four concepts from US based fast reactor companies.

2. GENERAL ATOMICS-EM^2

2.1. Reactor plant parameters [3–5]

Name: Energy Multiplier Module (EM^2)
Power: 265 MW(e)
Coolant: Helium gas
Fuel type: Uranium carbide kernel in silicon carbide cladding
Enrichment: <20% for starter fuel

According to General Atomics, the EM^2 takes advantage of high operating temperatures to achieve a high generating efficiency and small size to significantly reduce power costs. It achieves this through an all ceramic, high power density core with a game changing compact high speed turbine generator.

EM^2 embodies a direct Brayton cycle with an organic Rankine bottoming cycle to achieve net efficiency of 53%. The concept eliminates many of the systems and components normally associated with a conventional nuclear plant, thus reducing both the overnight capital costs and construction schedule. The compact plant is well suited to modularization, thereby enabling a greater portion of the plant assembly to be carried out in a factory and transported by truck, rail or barge to the deployment site.

Use of a fast spectrum combined with low neutron absorbing materials enables a convert and burn core with a 30-year fuel life without reshuffling or refuelling. The conversion ratio is 1.05 so that the core can be recycled with replenishment of only the fertile material. This eliminates the need for uranium enrichment after the first core. This significantly improves fuel utilization and reduced proliferation risk compared to existing technology.

The fuel cycle is closed through the use of a dry gas extraction recycling process, which avoids separating and extracting heavy metals. The waste stream is just fission products, which have no proliferation value and represent only 3% of the waste stream mass of current once-through reactors.

Additional features include improved safety features and a dry cooling option, which significantly expands siting options.

2.2. Why a fast neutron spectrum based reactor?

EM^2 is a compact, helium cooled, fast reactor that provides 265 MW(e) at economically competitive power rates. In order to reach the targeted energy densities with a compact modular system, a fast reactor design was selected.

2.3. Market objective and target market(s)

The target market for this reactor is worldwide electricity generation.

2.4. Uniqueness

EM^2 overcomes major performance and cost barriers associated with current reactors, namely, limited efficiency and scale, to enable nuclear power generation to be much more economically competitive within the worldwide energy mix.

2.5. Development/commercialization/funding

General Atomics has made uranium carbide kernels and fuel compacts of the desired stoichiometry. It has also made significant progress in silicon carbide cladding production, including the joining of silicon carbide materials. General Atomics has also developed and tested a high speed permanent magnet generator and designed the gas turbine and high efficiency inverter, which enables the power conversion unit to operate at the most efficient speed, independent of grid frequency. As the risks involved in creating these capabilities are retired, General Atomics expects to attract the necessary funding for programme completion.

2.6. Biggest challenge limiting the advancement of concept

The most challenging technical risks are being addressed through the development and testing of the fuel, ceramic core internal materials and the high speed power conversion unit.

3. GEN4ENERGY [6, 7]

3.1. General reactor plant parameters

Name: Gen4 module
Power: 70 MW(th), 25 MW(e)
Coolant: Lead–bismuth eutectic
Fuel type: Uranium nitride
Enrichment: 19.75%

3.2. Why a fast neutron spectrum based reactor?

A fast spectrum allows long core life duration (10 years) and neutronic simplicity.

3.3. Market objective and target market(s)

This reactor is specifically designed to replace diesel electricity generation in remote locations. Today, about 5% of worldwide electricity is generated using diesel or oil. This is very expensive power, particularly in remote locations. The Gen4 module provides a significant cost advantage versus diesel generation.

Specific target markets include remote communities that are difficult to connect to a major electrical grid (northern regions, islands, desert areas, geographically isolated areas), resource extraction activities including mining and oil and gas (typically large power consumers located in remote areas), and military facilities (facilities that require their own reliable power source for security reasons).

Rather than competing with large nuclear, coal, gas, or renewables, the Gen4 Energy concept competes against the most expensive power generation — diesel.

3.4. Uniqueness

Technologies were chosen to optimize the design for remote areas.

Reactor size: 1.5 m diameter × 2.5 m length — to fit in a spent fuel transportation cask.

Lead–bismuth coolant — non-flammable liquid metal coolant eliminates the pathways for radioactive contaminant release to a remote environment (versus sodium or water).

10-year core life, fuelled transport, and complete unit replacement — to avoid any on-site refuelling or reactor vessel entry in remote locations.

Fast spectrum reactor — provides operational simplicity for a smaller on-site staff.

3.5. Development/commercialization/funding

Funding to date (2007–2012) has been venture capital with no government support. Future funding is likely to be a combination of large sovereign wealth funds, governments and end users.

3.6. Biggest challenge limiting the advancement of concept

Funding of the required design and licensing process (estimated at US $300–500 million for Gen4) is the biggest challenge. The second biggest challenge is acceptance by the established nuclear community of a different technology (lead–bismuth coolant).

4. LAKECHIME

4.1. Reactor plant parameters [8, 9]

Name: USA-SVBR-100
Power: 100 MW(e)
Coolant: Lead–bismuth
Fuel type: Uranium oxide
Enrichment: Not reported

The design concept benefits from early conceptual design as well as materials science work performed by the US national laboratories and universities while leveraging extensive design and operating experience from collaborators in the Russian Federation. The proposed power output is 100 MW(e). The initial fuel is uranium oxide, a well understood fuel for liquid metal cooled fast reactor applications. Research and development is under way internationally on the use of mixed oxide as well as nitride fuel as follow-on options.

4.2. Why a fast neutron spectrum based reactor?

Fast reactors offer greatly enhanced sustainability by virtue of their ability to utilize natural and depleted uranium as well as components (uranium, plutonium and minor actinides) from LWR spent nuclear fuel. In addition, the USA-SVBR-100 utilizes a chemically inert coolant that enables passively safe processes to generate electricity.

4.3. Market objective and target market(s)

lakeChime and partners' goal is to provide a reasonable share of the global electric power supply and to primarily serve consumers in remote locations and in places with less developed infrastructure.

4.4. Uniqueness

Through collaborative effort based on proven technology (i.e. Russian experience with lead–bismuth cooled reactors for naval propulsion), the system could be fast tracked and advanced to construction and operation of a demonstration unit leading to early deployment.

4.5. Development/commercialization/funding

lakeChime anticipates receiving US Government funding and other international support.

4.6. Biggest challenge limiting the advancement of concept

National administrative obstacles of the collaborative partners are limiting factors.

5. TERRAPOWER

5.1. Reactor plant parameters

Name: Travelling wave reactor (TWR)
Power: 500–600 MW(e)
Coolant: Liquid metal
Fuel type: Metallic fuel
Enrichment: Less than 20% enriched driver fuel

The TWR concept is a high power density, liquid metal cooled fast breeder reactor. The first deployment will be a 500–600 MW(e) prototype design capable of qualifying the fuel and material concepts for the commercial version, which will have a power of approximately 1100 MW(e). All components for the prototype will be full size and designed for the larger commercial version. The balance of plant is a typical utility type Rankine cycle steam–electric plant utilizing a 3600 rpm turbine, with a conversion efficiency of approximately 42%.

The TWR deployment requires a metallic driver fuel enriched to less than 20%; the balance of the core is fuelled with depleted uranium. The driver fuel constitutes about 5% by weight of the core and the balance is depleted metallic uranium. The typical cycle is a breed and burn concept, where the end-of life core could be remanufactured to fuel a follow-on reactor without the need for enrichment or reprocessing, using additional depleted uranium.

5.2. Why a fast neutron spectrum based reactor?

The TWR was born out of the desire to mitigate worldwide poverty by providing safe, reliable, economical, base loaded electrical energy without contributing to greenhouse gas concentrations in the earth's atmosphere. Additionally, since nuclear energy is the only logical conclusion to achieve that goal on the scale required, the concept must also mitigate the usual issues associated with nuclear power, i.e. proliferation, safety and nuclear waste.

5.3. Market objective and target market(s)

Not stated.

5.4. Uniqueness

TWRs are targeting a 30-fold increase in fuel utilization efficiency when compared to conventional LWRs. They will also:

- Have significantly *enhanced safety* features that rely on the inherent physics of the design for automatic shutdown and passive cooling, even in the event of an accident that causes complete loss of both on-site and off-site power;
- Be *cost competitive* due to greatly improved fuel utilization, reduced uranium mining and fuel purchases, reduced need for enrichment facilities, elimination of costly reprocessing plants, and lower costs for waste transportation and disposal;
- Be *environmentally friendly* by using waste depleted uranium as its main fuel, producing less waste than current reactors, and reducing transportation and disposal requirements;
- Be *resistant to weapons proliferation* since the nuclear vessel can remain sealed and can operate for up to 40 years without changing out fuel assemblies; the TWR minimizes and ultimately eliminates the need for uranium enrichment; and no chemical reprocessing is required; and
- Greatly *enhance energy security* since less uranium is required, making plant owners and countries far less vulnerable to uranium supply shortages or disruption, and commodity price increases.

5.5. Development/commercialization/funding

TerraPower's global working relationships and analytical tools have enabled the company to develop a conceptual design of a nuclear plant that will meet all of the design objectives. The company has had extensive negotiations with institutions around the world and is now preparing to:

— Test fuel designs in Russian and US reactors;
— Establish a partnership with a host government and its companies;
— Identify a site and funding for a prototype facility; and
— Complete the design, engineering and construction of the prototype facility.

TerraPower has a detailed plan to complete the development of the TWR technology and bring it to a state of commercial readiness by around 2022.

6. SUMMARY TABLE

Table 1 is derived from information provided by the authors.

7. ANALYSIS

All of the reactor concepts and designs proposed in this paper intend to meet or exceed Gen IV standards for safety. Liquid metal cooled fast reactors have the benefit of operating at near atmospheric pressures. This provides both safety and cost benefits. Safety benefits accrue since the reactor cores are not highly pressurized thereby reducing risks from leakages or explosions. Cost benefits accrue since reactor vessels do not require the additional vessel thicknesses and pressure equipment necessary to maintain high interior pressures. Liquid metal reactors can also be designed to self-modulate when power demands are placed on them. This potentially makes these good candidates for use in remote locations or where a large skilled operating force is difficult to sustain.

The Gen4Energy concept is the smallest of the four reactors, so small that the entire reactor vessel can fit inside a recycled fuel transportation cask. Transportation issues are thereby minimized. This reactor is intended to function almost like a nuclear battery that can be shipped to a site, used for a long lifetime (~10 years), cooled perhaps for several more years, and then returned for fuel disposition. A sealed core and long lifetime dramatically reduce proliferation risk. Minimal operator intervention is required to maintain this system from a power perspective. The target market is the Arctic, where the primary fuel source is diesel. Although a US based reactor licence is strongly desired, continental US sites for this reactor are limited. Hence, the company is open to international licensing and investment opportunities from countries with a higher percentage of its population residing in remote (diesel fuelled) destinations. Outside of

TABLE 1. COMPARATIVE SUMMARY OF FOUR FAST REACTOR CONCEPTS

Organization	Design concept name	Power output per module (MW(e))	Fuel type	Coolant	Planned core lifetime (years)	Target market	Planned funding source	Unique design challenges	Key advantages
General Atomics	Energy Multiplier Module (EM^2)	265	All ceramic	Gas (He)	30	Worldwide electricity generation	To be determined	Silicon carbide cladding development High speed permanent magnet generator	High efficiency high temperature direct Brayton with organic Rankine bottoming cycle power
Gen4Energy	Gen4 module	25	Uranium nitride	Lead–bismuth eutectic	10+	Remote sites currently powered by diesel fuel, primarily non-US	Venture	Funding for licensing and acceptance of lead–bismuth cooling by nuclear community	Sealed power module fits inside a single spent nuclear fuel shipping cask
lakeChime	USA-SVBR-100	100		Lead–bismuth		First design in non-US market	Government funding and others		Based on existing, proven Russian designs
TerraPower	Travelling wave reactor (TWR-P)	500–600	Metallic	Liquid metal	40	High energy demand markets	Venture	Materials/fuel	High efficiency, high power density, depleted U fuel

licensing, this reactor has perhaps the fewest technical challenges towards achieving deployment of the four companies represented in this paper. The market is well defined and there is no shortage of potential customers for a nuclear battery scenario.

The TerraPower TWR also intends to take advantage of the sealed vessel for proliferation resistance but is designed to be a major power source capable of providing power that matches what a typical LWR can generate today. An important intended characteristic of this reactor is the extremely high (~30%) fuel burnup that can be achieved by shuffling the fuel rods to maximize re-burning of the actinides that are created. In effect, core lifetimes could be as long as 40 years, greatly reducing the cost of fuel to maintain these reactors, and the corresponding waste, even without recyling. In the USA, sodium has an advantage as a coolant over lead (and lead–bismuth), at least as far as licensing goes, since the USA has a long history of data and experience acquired with sodium. Unfortunately, the US political climate is difficult enough that (even with solid venture capital behind the reactor) the company is seeking licensing and deployment in countries such as China and the Republic of Korea. This is the greenest of the four reactor scenarios and can even utilize a large fraction of spent nuclear fuel in its initial fuelling. Hence, it has the advantage of addressing the current nuclear waste problem, even if there is no reprocessing of the TWR spent fuel. There are important technical challenges ahead, particularly in the area of achieving high burnups. Cladding and materials will also be important considerations. In addition, since fuel is not the main expense in most reactors today, the business model relies on keeping the costs of building the reactor as low as possible, utilizing all the compact modularity and high energy density benefits of fast reactors. It is anticipated that these costs should be below the cost of a comparably sized LWR. Potential customers may also be willing to pay a premium for this reactor, owing to its ‘greenness’.

The lakeChime concept is unique among these four companies since it plans to collaborate and assist with certain aspects of the Russian SVBR-100 [10] that is under planned development for deployment around 2020. In particular, some of the aspects include control designs, international safety protocols, non-proliferation aspects and eventual US Nuclear Regulatory Commission licensing. This reactor design has been highly optimized for compactness and modularity cost benefits and there is a long history of lead based coolant data and experience acquired in the Russian Federation. The core is not sealed as in the other reactors described since the SVBR-100 is being designed to operate in a closed fuel cycle. Proliferation risks are reduced slightly due to the length of fuelling cycle being approximately 8 years, thereby reducing the handling of the spent fuel. Even a once-through fuel cycle in these reactors could still be cost effective, although fuel burnup is not nearly as high as anticipated for the

TWR. The company, lakeChime assumes minimal risk since its role is to serve primarily as a partner and facilitator.

The General Atomics EM^2 approach is different from each of the other vendors. Instead of a liquid metal coolant with the potential liabilities of steel corrosion in the case of lead and fire in the case of sodium, the EM^2 plans to use the inert noble gas helium. Helium does not activate and has no problematic interactions with the environment. Another planned advantage for the EM^2 is the direct drive Brayton cycle power generation module. The direct drive mechanism eliminates costly secondary loops that are required in nearly all reactor designs, both fast reactor and LWR. To make this cycle even more effective, temperatures are specified to be of the order of 900°C. By putting all these concepts together into a compact modular design, and combining these into paired configurations of eight or more reactors to provide power in the 2 GW range, General Atomics is targeting a cost reduction of conventional nuclear power by 50% or so. Potentially, that could make this design cost compatible with natural gas plants, depending of course on where the market for natural gas finally settles down. General Atomic's focus is towards gaining a US Nuclear Regulatory Commission licence that would enable this product to be marketed in the USA and elsewhere. Additionally, General Atomics is proposing a fuel recycling process that targets fission product separation instead of actinide separation to reduce risks of proliferation. If this is proven to be effective, the fuel cycle could be closed and existing nuclear waste could be burned in these reactors to help address the waste problem. Of the group, the EM^2 concept has the highest potential reward, but it also has the highest set of technical risks as it unabashedly pushes beyond the range of known material properties. It will require new material research and designs for cladding and fuel that can withstand the high temperatures and pressures required to operate this reactor. Funding for continued research in these areas will need to come largely from the US Government.

8. CONCLUSIONS

While a considerable amount of development and testing will be required to bring to deployment the four designs discussed in this paper, entrepreneurs are working on modular reactor power plants based on a variety of business models and promising technologies. 'Thumbnail' summaries of the four models have been presented in this paper. Each offers promise to satisfy a particular application and market where cost reduction is a primary factor. Funding remains an ongoing challenge for each of these companies and it will take time for the international marketplace to sort through all of the creative entrepreneurial ideas.

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FAST REACTOR OPERATION AND DECOMMISSIONING: INTERNATIONAL EXPERIENCE

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PROTOTYPE AND INDUSTRIAL SFRs: YESTERDAY, TODAY AND TOMORROW

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Abstract

This paper first provides an overview of prototype and industrial sodium cooled fast reactors (SFRs) before describing operating experience for availability and for safety. Two important topics will also be discussed: the fuel cycle for these reactors and their dismantling. The main conclusions drawn by EDF as a potential investor, industrial architect and operator of future SFRs will then be presented. Efforts towards improvements must focus firstly on availability (duration of programme shutdowns, reliability, materials, etc.); safety objectives are now stricter than they were in the past (core meltdown accidents without external consequences, etc.) and finally, costs for the SFR and its fuel cycle must be brought up to the same level as those for the most economical electricity generation means.

1. INTRODUCTION

As early as in 1945, Enrico Fermi declared at Los Alamos, “The first country to develop a fast breeder reactor will have a commercial advantage for the exploitation of nuclear energy”. Even so, in 1956, Hyman Rickover believed that these reactors were “expensive to build, complex to operate, susceptible to prolonged shutdown as a result of even minor malfunctions, and difficult and time-consuming to repair”. Three generations of researchers, engineers and nuclear reactors later, the world situation is highly contrasted in countries that have attempted to develop fast breeder reactors: a few have stopped, others are continuing, some have begun again, while some are steering towards new types of reactor, particularly with different coolant fluids.

The sole concern in this presentation is sodium cooled fast reactors (SFRs) designed primarily for electricity generation, even if only as prototypes. For EDF, as a potential investor, industrial architect and operator of future 4th generation reactors to be used for electricity generation, when these reactors become necessary and competitive, all information from the past needs to be thoroughly analysed, considering details of how these reactors function, their fuel cycle and their dismantling, in order to steer R&D, and then design, towards the most robust options with the best likelihood of satisfying the needs of the electricity producer.

2. PROTOTYPE AND INDUSTRIAL SFRs

Table 1 lists the various prototype and industrial SFRs (excluding experimental reactors) under deconstruction, in operation or under construction (note that the concepts of experimental, demonstrator, prototype or industrial reactor are subject to debate; the following classification only represents the opinion of the author).

At the moment, there is a total accumulated genuine experience of about 105 years with prototype and industrial SFRs (apart from long shutdowns). This value is not negligible, but nevertheless is modest compared with accumulated operating experience with industrial pressurized water reactors (more than 9000 years) and boiling water reactors (more than 2500 years). Total electricity generation is now more than 150 TW·h, two-thirds of which is from the Russian BN-600 reactor.

3. DESIGN OF SFRs

An examination of the design of SFRs throughout the world with reference to publications and networks for information exchange between operators shows relatively similar design solutions, even though design choices were made at a time characterized by competition between States, nuclear R&D agencies, designers and manufacturers. When the time came for cooperation, or at the least exchanges between operators about their operating experience, these similarities facilitated reciprocal understanding between them [1–8].

Nevertheless, there are important differences between these reactors that are still true to the present day, and that are compared, sometimes intensively, to define design choices for future SFRs. This is why it is important to mention some of the main differences here.

SFRs included in this presentation use an oxide type fuel (highly enriched uranium oxide or uranium and plutonium oxide) that is also used predominantly in 2nd and 3rd generation water reactors, except for (large) differences in the contents of fissile materials, densities, operating temperatures and manufacturing methods. Nevertheless, other fuel types were used in experimental SFRs or are envisaged for 4th generation SFRs: metal (high breeding gain, increased thermal conductivity), carbide and nitride (limited interaction with sodium, margin for fusion of fuel).

The primary circuit is based on a pool type reactor (Phénix, PFR, BN-600, Superphénix, BN-800, PFBR). All, or almost all, of the primary sodium is contained in a vessel inside which the various components (pumps, intermediate heat exchangers, etc.) are immersed. Otherwise, and more commonly, the

TABLE 1. PROTOTYPE AND INDUSTRIAL SFRs UNDER DECONSTRUCTION, IN OPERATION OR UNDER CONSTRUCTION

Reactor (country)	Thermal power (MW(th))	Electrical power (MW(e))	First network coupling	Final shutdown	Operating experience (years)	Electrical production (TW·h)
Fermi 1 (United States of America)	200	65	5 Aug. 1966	29 Oct. 1972	2.5[a]	0.03
BN-350 (Kazakhstan)	750	135+ desalination	16 Jul. 1973	22 Apr. 1999	24.5[b]	1.9
Phénix (France)	563 (then 350)	250 (then 142)	13 Dec. 1973	6 Mar. 2009	23.5[c]	24.4
PFR (United Kingdom)	650	250	10 Jan. 1975	31 Mar. 1994	19	7.1
BN-600 (Russian Federation)	1470	600	8 Apr. 1980		33	112.1 (2011)
Superphénix (France)	3000	1200	14 Jan. 1986	24 Dec. 1996	4.5[d]	7.9
Monju (Japan)	714	280	29 Aug.1995		0.5[e]	

TABLE 1. PROTOTYPE AND INDUSTRIAL SFRs UNDER DECONSTRUCTION, IN OPERATION OR UNDER CONSTRUCTION (cont.)

Reactor (country)	Thermal power (MW(th))	Electrical power (MW(e))	First network coupling	Final shutdown	Operating experience (years)	Electrical production (TW·h)
BN-800 (Russian Federation)	2100	800	2014?			
PFBR (India)	1250	500	2014?			

Note: Green: reactors in operation; yellow: reactors under construction.

[a] Fermi 1 was shut down from October 1966 to July 1970 due to the partial meltdown of two subassemblies.

[b] BN-350 hardly operated in 1973 and 1974 (and not very much in 1975) owing to sodium–water reactions in the steam generators.

[c] Phénix was shut down from September 1990 to February 1993 and from March 1993 to December 1994 (emergency shutdowns due to negative reactivity) and then from April 1995 to May 1998 and from December 1998 to June 2003 (safety re-evaluation and renovation work).

[d] Superphénix was shutdown from May 1987 to March 1989 (sodium leak in the external interim storage tank) and then from July 1990 to August 1994 (pollution of primary sodium, then work to strengthen protection against sodium fires) and from December 1994 to July 1995 (argon leak in an intermediate heat exchanger).

[e] Monju was shutdown from December 1995 to May 2010 (secondary sodium leak, followed by reinforcement work) and from August 2010 (the fuel handling device was dropped).

primary circuit is based on a loop type reactor (Fermi 1, BN-350, Monju). The former are characterized by a higher thermal inertia and simple confinement of the radioactive sodium, while the latter offer particularly better resistance to very strong earthquakes, are more easily inspectable with respect to internal structures and could eventually lead to an SFR design with no intermediate circuits. The CEA, AREVA and EDF have also considered a hybrid concept (with loops, but with the primary pumps immersed in the reactor block) as part of long term R&D actions.

Steam generators are some of the most sensitive elements of SFRs because they are an essential link in energy conversion between the core and the turbine. They are also the main component in which the risk of a violent reaction between sodium and water can occur. In general, designers have the choice between modular or single-piece steam generators (the main selection criteria are the maturity of fabrication and safety), and between different concepts: straight tubes or spiral-wound tubes, expansion by expansion loops or bellows, tubes containing water or sodium, etc. Many of these concepts have never been manufactured and there is no operating experience to form the basis for preferring one or another. Note that the CEA is studying an energy conversion system for the ASTRID prototype reactor based on a gas turbine (nitrogen) as an alternative to the traditional water–steam cycle, to exclude the risk of sodium–water reactions.

Finally, the comparison made by the JAEA and EDF between the JSFR reactor project and the EDF specifications for future SFRs (see Ref. [9]) has shown that several differences between SFRs are due to the fact that safety requirements fixed in each country for these reactors are not yet stable and have not been harmonized (and are too often inspired by requirements for LWRs for them to be used industrially). This is quite understandable for a reactor type that can unashamedly be qualified as young.

4. AVAILABILITY OF SFRs

Table 2 presents, firstly, the ‘gross’ value of the SFR availability ratio (K_d = energy produced/nominal power × time), and secondly, when appropriate, a more representative value of the availability of these reactors, ignoring particularly long reactor shutdown periods. The Fermi 1 and Monju reactors are no longer included, considering their low functioning time under power.

TABLE 2. SFR AVAILABILITY RATIOS

Reactor	K_d gross	'Corrected' availability ratio
BN-350	60%	
Phénix	34%	50–60% (apart from exceptional shutdowns)
PFR	21%	~ 40% (after corrections of steam generator problems in the first decade)
BN-600	73%	
Superphénix	7%	35–50% (considering authorizations from the Nuclear Safety Authority)

The net result for SFRs is contrasted, availability factors are not as good as would be expected for production units, but the results are sufficiently acceptable, considering that they are mainly prototypes, for this system to be considered for 4th generation production reactors (see Ref. [7]). In particular, the uninterrupted operating times of Phénix (151 days) and BN 600 (165 days) should be noted, although they were handicapped by their fuel renewal frequency. Immediately before the government decision to finally shut it down in 1996, Superphénix was coupled to the electricity network for 95% of the time, apart from programmed shutdowns. Similarly, the last automatic shutdown of BN-600 was in 2000 (and was due to a failure of the electricity network). Nevertheless, significant changes are necessary to improve the industrial nature of SFRs and some of these changes may represent a major change to what has occurred in the past.

Steam generators have been an important source of concern and unavailability on almost all studied installations, and these problems have significantly affected availability rates. The information to be retained is that steam generators (and auxiliary systems) must be designed and manufactured to be very reliable. They must especially be very easy to repair (or replace), easy to evaluate (in order to reveal the phenomenon that caused the leak at the origin of the sodium–water reaction) and easy to inspect.

In the future, improvements aiming at the availability of SFRs should be found to solve the problem of aerosol deposits that can prevent part movements (in blanket gas zones) and caustic stress corrosion control (choice of materials, design of parts, appropriate procedures, post-welding relaxation heat treatments, etc.), as well as reducing the occurrence of sodium leakages.

Reducing the numbers (and obviously the durations) of scheduled shutdowns is a good approach towards increasing the availability ratio, obviously at the same time as increasing the possibilities of maintenance during operation. In particular, this depends on an efficient and robust system for handling fuel assemblies.

Finally, reducing the number of control rods and significantly extending their service life can achieve many savings (times to change between operating ⇔ handling configurations, re-qualification times for shutdown systems, maintenance volume for rod mechanisms, etc.).

5. OPERATING EXPERIENCE FROM INCIDENTS

A nuclear reactor is operated during long periods without any incidents, described by a few values in terms of electricity generation or operating time. An incident occurs from time to time and is followed up by copious documentation and sometimes many comments. Talking about operating experience with a nuclear reactor often means focusing on these incidents, at the risk of suggesting that they are the only aspect of nuclear power. This is far from being the case. Nevertheless, the first step in improving the safety of facilities is to analyse these incidents. This is why a blacklist of these incidents has to be drawn up.

Although comparisons in the subject are difficult, it can be said that the number and nature of incidents that occur in SFRs are no different from the number and nature of incidents that occur with other reactor systems (although the equivalent of the Chernobyl, Fukushima, and even Three Mile Island accidents have never occurred). The most significant incidents in the history of all SFRs (experimental, prototype, industrial) apply to the following (in order of increasing safety importance):

— Sodium leakage (on average one every year and per reactor in operation, but the last leak declared at BN-600 occurred in May 1994) quantities are usually small (about one kilogram), they are quickly detected and do not generate any major fire; the largest leaks spilled a few hundred kilograms of sodium (KNK II in February 1968, BN-600 in October 1990, October 1993 and May 1994, Monju in December 1995).

— A sodium fire with modest amplitude, although it did cause damage to equipment in the room involved, diffused sodium aerosols into other rooms because the pipe concerned had not been emptied and ventilation had not been stopped (Monju in December 1995); there was also the spray sodium fire (about a dozen tonnes) that occurred in another type of facility using sodium as coolant (Almeria solar power station in August 1986).

— Leaks in steam generator tubes creating a weak sodium–water reaction (five leaks at Phénix, a dozen at BN-600, about forty at PFR) to violent (BN-350 in October 1973 and in February 1975, PFR in February 1987).
— Primary sodium pollution (Superphénix in June 1990, due to air ingress into the blanket argon circuit, PFR in June 1991 following an oil spill in the primary circuit).
— Cracks in some steels in contact with sodium and under specific conditions (15D3 steel at Superphénix in March 1987 and SNR 300, 321 steel at Phénix and PFR).
— Drop of a block of sodium impurities formed under the roof of the reactor into the primary circuit (BN 600 in January 1987), generating neutron and hydraulic disturbances.
— Damage to a fuel subassembly (FBTR in May 1987) or experimental equipment in the core (Joyo in June 2007), following rotation of the rotating plug.
— Blockages of control rod mechanisms by more or less oxidized sodium aerosol deposits (PFR between 1985 and 1987, KNK II in December 1986, December 1988, January 1991, Phénix in January 1974, March 1981, February 1985, July 1987, October 1987).
— Several variations in the reactivity, from various sources (EBR I in November 1955, DFR in November 1959, EBR II in October 1974, Rapsodie in 1978, KNK II in 1978, BN-600 in January 1987, Phénix in August 1989, September 1989 and September 1990, FBTR in November 1994 and during subsequent years).
— Partial core meltdown (Fermi 1 in October 1966) due to a metal plate that detached from its support and blocked sodium circulation in several subassemblies[1].

The only incident that can be qualified as an accident (in the sense used in nuclear safety) is the final point in the list, from which operating experience was drawn (particularly resulting in sodium supply to subassemblies through several side holes in their stands). Most of the other incidents above had no consequences on reactor safety, but some revealed weaknesses in the previous safety demonstration. Their analysis is particularly useful to draw up lines of R&D work described below to prepare the ground for future SFRs:

[1] A meltdown of half the EBR I core also occurred in November 1955 during a primary pump shutdown test without control rod drop.

— Innovative measures capable of preventing or slowing or jamming the control rod mechanisms, particularly due to the accumulation of sodium aerosols (more generally in the upper part of the reactor block) and diversification to prevent common mode incidents;
— A core design preventing or at least minimizing possibilities of compaction that could lead to a significant insertion of reactivity;
— Design of primary circuits to limit or even prevent significant volumes of gas (and vapourizable fluid or fluid reacting with sodium) that might be present or that could enter them, and monitoring and protection devices appropriate for the risks that they induce;
— Qualification programmes for new materials, taking account of their various conditions of use;
— Measures for prevention and monitoring of failures in the core support resistance;
— Devices for monitoring handling operations in sodium;
— The need and means for the complete unloading of the core;
— Methods of detecting firstly sodium leaks and secondly leaks from a steam generator tube, to help identify these events throughout their entire range of occurrence.

6. FUEL CYCLE

SFRs are associated with an almost completely closed fuel cycle to satisfy durability objectives[2]; this requires a robust, fluid and optimized nuclear materials circuit. Thus, the transport and processing of spent fuels must take place smoothly so that the fuel subassembly fabrication step is supplied with processed plutonium and uranium. Consequently, all steps in the 'closed' cycle are decisive to guarantee secure reactor procurement. This is not the case in the current cycle of the LWR fuel ('open') cycle in which the only crucial steps are the fabrication of new fuel. Furthermore, there is less operating experience for these links in this SFR fuel cycle (fabrication, processing, transport) than for other reactors. However, the 'closed' cycle makes reactors completely independent for the procurement of 'raw' materials, once stocks of uranium-238 (residues from enrichment) and plutonium (derived from spent fuel) have been built up.

[2] However, the cycle must be supplied with ^{238}U and it generates nuclear waste (fission products, etc.).

Fuel fabrication workshops have been constructed in every country that has built and operated SFRs. SFR fuels are usually composed of a dense bundle of rods contained in a hexagonal tube. They contain three times less nuclear material than a PWR assembly, but it is concentrated over a portion of the total height. Thus, considering the burnup rates and the efficiencies of each type of reactor, an SFR fuel fabrication plant must produce an assembly flow one and a half times higher than a PWR fuel fabrication plant, for equivalent electricity generation.

Finally, plutonium contents are two to three times as high as in PWR MOX assemblies, which increases constraints related to temperature, criticality and radiation shielding. French operating experience shows satisfactory functioning of the process for fabrication of Phénix and Superphénix fuel pellets, which is technically simpler than the process used at the MÉLOX plant for PWR MOX fuels. However, there are some difficulties with welding operations of assembly components. Assuming that the future French fleet is composed solely of SFRs, it would also be necessary to monitor the secure procurement of the fuel supply (advantage of having several fabrication plants) and exposure of staff working in these plants and during the transport of subassemblies (by virtue of the ALARA principle and taking account of probable reductions in regulatory limits).

SFR spent fuel subassemblies are very different from PWR assemblies, both in terms of their mechanical structure and their components, owing to their high burnup (they contain more platinoids, more fission products, more plutonium and more minor actinides per tonne of oxide). The effects of restructuring the fuel ceramic lead to a strong release of fission gases and internal corrosion of the jacket, which is also subject to high fluences. Although the treatment of spent fuel subassemblies from Rapsodie and Phénix (several tens of tonnes) has not approached industrial conditions at a rate of up to a few tonnes per year, it has revealed the main difficulties and demonstrated feasibility.

A new design should be proposed for future workshops in the SFR spent fuel treatment plant, different from the water reactor fuel treatment plant (La Hague in France). This is particularly necessary for reception, the process pilot (dismantling of hexagonal tubes, shearing) and treatment of structural waste (flow three to four times greater than at the present time). To a lesser extent, this is also true for dissolution, separation by liquid–liquid extraction and plutonium workshops. Assuming a future French fleet composed solely of SFRs, it would also be necessary to monitor the capacity of the treatment plant, which should produce seven times more plutonium than La Hague at the moment, which, a priori, would make continuous operation essential. Its reliability would also require monitoring because it guarantees the procurement of the cycle with fissile materials.

Finally, the transport of new or spent SFR fuel subassemblies is particularly important; rotations in just-in-time flow are necessary, whenever possible, to optimize the management of nuclear materials both economically and in terms of dosimetry. SFR subassemblies are characterized by local high source terms (power, neutron and gamma sources). This requires, as a minimum, that decay heat removal capabilities be improved, biological shielding for transport packaging be reinforced, and changes to regulations for transport of fissile nuclear materials also be considered.

Although the first studies confirm that it is feasible to transport several spent subassemblies in a single cask after several years cooling while respecting temperature and radiation shielding criteria, it is possible that studies on safety, criticality, mechanical strength, confinement of materials (particularly in accident conditions) lead to changing the feasibility of transporting complete spent SFR subassemblies. Disassembly in a hot cell close to reactors would then be necessary, which would form a very strong industrial constraint and would significantly increase costs.

7. DECONSTRUCTION

Only Fermi 1 and Superphénix among the prototype and industrial reactors described in this paper have reached a significant stage of dismantling. Nevertheless, deconstruction of small experimental SFRs also provides information, and this is why they will also be included below. Table 3 summarizes the state of dismantling of all SFRs.

TABLE 3. DISMANTLING STATES OF VARIOUS REACTORS

Reactor	Shutdown	Current state of dismantling and future activities
EBR I (USA)	1963	Installation cleaned and transformed into a museum
Fermi 1 (USA)	1972	Kept ‘mothballed’ after elimination of sodium and partial cleaning of the installations, reactor vessel and circuits under CO_2
DFR (UK)	1977	NaK drained and destroyed (not including retention tanks), there are still some fertile assemblies in the reactor

TABLE 3. DISMANTLING STATES OF VARIOUS REACTORS (cont.)

Reactor	Shutdown	Current state of dismantling and future activities
Rapsodie (France)	1983	Sodium drained and destroyed, reactor vessel isolated under nitrogen, preparation for dismantling
KNK II (Germany)	1991	Reactor vessel dismantled, dismantling of the reactor pit in progress, 'return to green site' planned for 2019
FFTF (USA)	1992	'Mothballed', sodium drained and destroyed, reactor vessel and circuits under argon
PFR (UK)	1994	Sodium drained and destroyed, preparation for the treatment of residual sodium retention tanks
EBR II (USA)	1994	Sodium drained and destroyed, carbonated residues, reactor vessel and circuits kept under CO_2 awaiting washing (2013)
Superphénix (France)	1998	Sodium drained and transformed into soda contained in concrete blocks, withdrawal and dismantling of large removable components
BN 350 (Kazakhstan)	1999	Assemblies unloaded, installation 'mothballed'
BR 10 (Russian Federation)	2002	Preparation for dismantling
Phénix (France)	2009	Withdrawal and dismantling of removable components, core unloading programmed for 2013 after performing an ultimate test

Only one major incident occurred; during the treatment of residual sodium in a Rapsodie storage tank by reaction with an alcohol, hydrogen was suddenly released and the tank exploded, killing the technician who was monitoring the operation and injuring another.

The main information that can be drawn from studies and operations related to deconstruction of these reactors, particularly related to the design of future SFRs, is as follows:

— Complete core unloading is a long operation which sometimes requires processes or equipment that were not considered in the operations phase.

— Complete draining of sodium from the reactor is another long operation that requires complex actions. Note that up to now, the future of sodium has been different in different installations (direct or indirect reuse, rejection of sodium salt into the river or sea environment, incorporation into concrete) depending on choices sometimes dictated by the regulations or even by specific considerations,
— The presence of sodium in the form of aerosol deposits (e.g. in penetrations through reactor top covers) makes flooding of the reactor with water more complex when this process is selected to provide biological shielding during dismantling operations.
— Cold traps (or similar equipment) in which sodium compounds (oxides, hydrides, etc.) and radioactive elements (activation products, fission products if there are any leaks in fuel cladding, etc.) concentrate during the reactor lifetime are the parts that generate the greatest chemical and radiological risks during dismantling (undoubtedly, final storage in the existing condition and with an adapted confinement would be preferable).
— Treatment of sodium-potassium (NaK) alloy, particularly when it has been oxidized, creates chemical risks that require the perfect control of a complex process.
— The radiological source term is concentrated in a few structures close to the core, particularly in the presence of some materials such as stellites that become very strongly activated under neutron flux (nonetheless, the global activity of nuclear waste derived from SFRs is lower than in other types of reactor since much of it can be sent to conventional waste disposal systems).
— Dismantling of components that have been in contact with primary sodium causes tritium release in proportions that are difficult to estimate (this can seriously slow down these operations depending on the nature of release authorizations).

In general, there is no technical blockage or major difficulty in the deconstruction of SFRs, in any case no more than for dismantling other types of nuclear reactor or installation.

8. REACTORS UNDER CONSTRUCTION OR PLANNED

Two SFR power plants are currently under construction, namely the BN-800 on the Beloyarsk site in the Russian Federation and the PFBR on the Kalpakkam site in India. In China, the CEFR experimental reactor on the Tuoli site (65 MW(th), 23 MW(e)) was connected to the electric grid for one day in July 2011.

The Russian Government has decided to construct the BN-1200 reactor, also on the Beloyarsk site. India is planning the construction of six twinned CFBR 500 MW(e) reactors by 2025. It has been reported that China has bought two BN-800 reactors, and also locally designed projects. The Republic of Korea is also working on a 600 MW(e) SFR project. In Japan, the JSFR project (1500 MW(e)) has been put on hold after the Fukushima Daiichi accident. Finally, in 2010, France initiated studies of a reactor called ASTRID (600 MW(e)).

All of these projects are described in other conference presentations.

9. INFORMATION FROM THE PAST

This paper has provided technical information drawn from operation of the Phénix and Superphénix reactors and from other SFR rectors based on a publication presented at the FR09 conference (see Ref. [6]). This paper has also focused on improvements that are essential for EDF as an electricity producer so that eventually these reactors can become a competitive means of electricity generation (compared with LWRs and non-nuclear means, particularly gas, coal and renewable energies) with a degree of safety sufficient to satisfy the public (in other words, as a first analysis, at least equivalent to LWRs contemporary with them).

Firstly, improvement efforts must focus on the availability of SFRs. It has been seen that ‘corrected’ availability rates of previous reactors were between 50 and 75% and that they were sometimes capable of achieving rates of about 80%. If they are to be competitive with other production means, the availability rate will have to be more than 90%. Innovations in the field should aim, as a priority, at shutdown times for refuelling, maintenance and in-service inspection, extension of operating cycles and reliability of the components, particularly those on the energy conversion line between the core and the turbine (intermediate heat exchangers and circuits, steam generators, etc.). In particular, materials and their industrial use should be such that sodium leaks become rare and that the design of circuits and equipment enables their repair or replacement in a very short period (a few days).

In terms of safety, we now require a level equivalent to safety requirements fixed by the Safety Authorities Association (WENRA [10, 11]) plus lessons learned from the Fukushima Daiichi nuclear power plant accident. This latter information does not seem necessarily prohibitive for SFRs, but adequate prevention and mitigation measures will have to be set up to guarantee that the primary sodium inventory is maintained safe under extreme accident conditions (water will not spill into the primary circuit of an SFR even in the worst situation). It is nonetheless worth pointing out that well designed SFRs are particularly easy to cool in a degraded situation; there are large margins between normal operation and the boiling of sodium, circulation of sodium in natural convection and the atmospheric heat sink.

Considerable progress still has to be made in the various fields to improve the safety of future SFRs in comparison with SFRs that have been constructed up to now, and particularly:

- Reliability of the reactor shutdown function, which does not benefit from diversity added by soluble boron in a PWR;
- Knowledge and understanding of a core meltdown in order to determine the necessary and adequate mitigation measures;
- Qualification of materials and their use, concerning their life and their behaviour in the presence of sodium or a sodium compound (soda, etc.);
- Handling of fuel assemblies under sodium;
- Coherent treatment of the impact of accidental radioactive releases (nuclear accident) and chemical releases (sodium or sodium compounds).

Finally, considerable progress is also needed to reduce the costs of SFRs and their fuel cycle to make them competitive with other electricity generation means (nuclear, conventional or renewable) without waiting for uranium to reach prices of about US \$200–300 per pound of U_3O_8. Apart from a reduction in construction and operating costs (of the reactor, fuel fabrication and treatment plants, and also transport of new and spent subassemblies), efforts should be made on in-service inspection, reliability and life of equipment, maintenance and repair or replacement, without forgetting to plan dismantling operations, particularly by minimizing volumes and the activity of future waste.

Nevertheless, economic optimization is secondary until two major elements are acquired: firstly, the need for a more provident use of uranium resources (facing a risk of shortage) and secondly democratic and continuous trust of the population in this type of reactor and everything that it represents (long term nuclear energy).

10. CONCLUSION

The future of SFRs is not certain. It depends on the state of natural uranium resources and their consumption, and prospective studies show that the ‘uranium peak’ is not likely to be reached before the end of this century. However, SFRs could find a place alongside 2nd and 3rd generation reactors, with a view to optimizing management of nuclear materials. In all cases, construction of industrial SFRs will depend on the capacity of R&D agencies, designers, manufacturers and operators to make safe and long lasting SFRs that produce megawatt-hours competitive with other generation means regardless of whether these means are nuclear, conventional thermal or renewable.

In any case, the time needed to develop a new reactor system is very long [12]. In the past, it took about thirty years when the economic or strategic demand was strong and objections were weak. Consequently, we should not wait until we have our backs to the wall or we have reached the edge of a cliff before initiating and then constructing new reactor projects, while integrating all necessary innovation. Innovation is only relevant when it is based on the analysis of accumulated experience.

It will be useful to share our knowledge within the community of engineers, researchers and technicians working on SFRs, so that we can participate under optimum conditions. This is one of the merits of the IAEA, and also WANO and various learned societies such as the SFEN in France, in organizing these exchanges so that we can make progress.

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RECENT PROGRESS AND STATUS OF MONJU

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Abstract

Monju, Japan's prototype sodium cooled fast reactor of 280 MW(e) class, restarted its test operation in 2010 after a long stoppage since the sodium leak accident in 1995. The zero power system startup tests were successfully conducted. The major achievement of the tests was an accurate prediction of reactor physics parameters with a core having a complex fuel composition that includes americium-rich fuel. The hardware troubles recently experienced, none of them being safety significant, have been restored to make the plant ready for the next power increase tests. The reactor, however, has been put into a standby mode again since the Fukushima-Daiichi accident of 11 March 2011, at least until the national energy and environment programme is revised and a research plan using Monju is developed through 2013. We believe the roles of Monju will not change and comprise further enhancement of safety against severe accidents; demonstration of stable power generation and actinide burning; provision of technology and knowledge base for future sodium cooled fast reactors; and use of the plant as an international research facility.

1. INTRODUCTION

Monju, Japan's prototype sodium cooled fast reactor (SFR) of 280 MW(e) class, restarted its test operation in 2010 after a long stoppage since the sodium leak accident in 1995. The zero power system startup tests (SSTs) were successfully conducted after long reactor shutdown. The major achievement was accurate prediction of reactor physics parameters with a core having a complex fuel composition that includes americium-rich fuel. The hardware troubles recently experienced, none of them being safety significant, have been restored to make the plant ready for the next power increase tests. The reactor, however, has been put into a standby mode again since the Fukushima-Daiichi accident of 11 March 2011. This accident went into severe core damage progression due mainly to the loss of all the electric power (station blackout or SBO) and the loss of decay heat removal capability that are essential elements for reactor safety assurance. Similar to other light water reactors (LWRs) in Japan, safety improvement efforts have been implemented in Monju as well to prevent and to mitigate severe accident progression.

The energy and nuclear policy debates during 2012 have changed the environment of nuclear research and development (R&D) activities at the Japan Atomic Energy Agency (JAEA), including the future of Monju. A government level discussion is under way on the future direction of R&D using Monju. In this paper, the history and progress, current status and future prospects of Monju are described with emphasis on the safety features of Monju and post-Fukushima safety improvement.

2. HISTORICAL PERSPECTIVE AND ACHIEVEMENT

2.1. History of Monju

The history and present status of Monju is briefly explained. The safety review for a construction permit was finished in 1983, followed by plant construction and component manufacture, which were completed in 1991. The initial criticality was attained in 1994 and the series of SSTs was initiated. In December 1995, when the SST was conducted at the 40% power level, a sodium leak accident occurred in the secondary heat transport system, just outside the reactor containment vessel. The leaked sodium, amounting to 640 kg at most, was non-radioactive and did not affect the safety (reactor shutdown and cooling) of the plant and resulted in no environmental consequences. Nevertheless, the accident was treated very seriously by the mass media, and many people thought Monju must be dangerous because liquid metal sodium is dangerous. Further, in those days accident information was less openly disclosed compared with the present day, and this gave the public a negative impression on Monju. These are some of the major reasons why it took more than a decade before the reactor could be brought back to test operation. The cause of the sodium leak accident has been thoroughly investigated and the plant modified to reinforce the safety measures against sodium chemical reactions and an approval has been obtained from the regulatory authority and the local government. In May 2010, the first step of SSTs, zero power core confirmation test, was initiated after the long reactor shutdown period of more than 14 years.

Up until the restart of Monju in 2010, a technology base to design and construct SFRs has been established in Japan based on the experience acquired through the experimental reactor Joyo and a comprehensive research and development programme in the areas such as safety, fuel and material, components, thermohydraulics, and instrumentation and control. The technologies relevant to Monju have been further advanced, even after the sodium leak accident, with the activities that include safety improvement against sodium leakage and chemical reactions, back-check seismic safety evaluation, and other improvements in plant management system for operation and maintenance.

2.2. SSTs

The major achievement of the first step of restarted SSTs, the zero power core confirmation test, is highlighted by successful operation of the reactor and the cooling system without major trouble even after a long blank period with the reactor core containing americium-rich fuel. During the long period of stoppage, the plant operators have repeatedly taken a series of training and education courses, especially using the Monju plant simulator called the MARS system, to maintain and further improve their plant knowledge and operating skills. Through this SST, it has been possible to demonstrate the safe control and operation of the reactor and heat transport systems and confirm that the inherent negative reactivity feedback features are effective as designed. Although the core consisted of three different types of fuel subassembly containing americium-rich 14 year old fuel, it has been shown that the criticality and other reactor physics parameters are accurately predicted.

2.3. Recovery from hardware troubles

Monju has experienced some hardware troubles during the last few years. Although none of them are safety significant, they have now been restored to make the plant ready for the next power increase tests. Among them, the trouble of incidental drop within the reactor vessel of the heavy structure called an in-vessel transfer machine (IVTM) is briefly explained here.

The IVTM is a special device not used during plant normal operation but which is inserted into the reactor vessel for dedicated use in refuelling. It is a 12 m long, pipe-like structure weighing ~3 t. Using this, a spent fuel subassembly is exchanged for a fresh fuel subassembly. After the successful SST, the IVTM was inserted into the reactor vessel and part of the reactor core was refuelled. Then the IVTM was raised up through a sleeve structure, the machine dropped from a height of about 2 m to strike the vessel head plug structure. This drop turned out to be caused by the gripping structure of the lifting machine failing to latch on to the top of the IVTM completely. It took nearly a year to recover the dropped IVTM from the reactor vessel, because this required a careful and detailed work procedure to handle the heavy structure in a narrow space above the reactor vessel and not to allow in-vessel sodium coming into contact with air. The possibility of structural damage has been evaluated in detail and all the parts comprising the IVTM were fully recovered. Fortunately, it was concluded that the vessel plug and internal structures were not affected and remained intact.

The gripper structure of a lifting machine was modified and the damaged IVTM was replaced with a newly manufactured one. A test refuelling operation then demonstrated that this IVTM trouble was completely resolved.

3. POST-FUKUSHIMA SAFETY IMPROVEMENT

3.1. Safety characteristics of Monju

The accident at the Fukushima-Daiichi plants, which took place on 11 March 2011, has enormously affected the operation and management of all the nuclear power plants in Japan. Monju, a sodium cooled fast reactor having a different design, was no exception, even though the plant is still in the construction stage. The timing of the Fukushima accident coincided with recovering from the hardware troubles, such as the IVTM incident, and conducting a series of preparation efforts for the second step of SST, namely 40% power plant confirmation test, which includes the function testing of the balance of plant (a water and steam system and turbine).

It must be noted that Monju has some inherently advantageous safety features under a severe accident condition of SBO and the resultant loss of heat sink. First, the major plant facilities are built on a ground level 21 m above sea level and this is a significant advantage against tsunamis. Second, the ultimate heat sink for decay heat removal in Monju is 'air'. This is in contrast to LWRs elsewhere in Japan whose heat sink is seawater. Thus, the cooling system in Monju does not rely much on the seawater system. In addition, in the event of SBO, the decay heat from the core fuel can be transported through the sodium cooling system to the air cooler by 'natural circulation'. This is a passive, and hence extremely reliable, safety feature of sodium cooled reactor systems, and the effectiveness of natural circulation heat removal has been demonstrated by many reactor experiments conducted in the world.

3.2. Safety improvement measures

Similar to the other LWR plants in Japan, we have been asked to confirm the safety of Monju in the event of a huge earthquake and tsunami, and against the potential severe accident progression following the natural disaster, and to consider safety measures to provide additional safety margins beyond the design basis. Since this severe accident, we have reviewed and reinforced, as far as appropriate, the safety of Monju, taking full account of the safety design characteristics of the sodium cooled fast reactor system and the accident progression in Fukushima and lessons learned, as requested by the regulatory authority and the local government.

The hardware measures installed to improve the safety under the SBO condition include deployment of high voltage power vehicles, preparation of a backup seawater pump, watertightening of seawater piping into plant buildings, provision of multiple communication systems, including satellite phones, improved ventilation and air conditioning for the main control room and radiation protection equipment. New hardware requires a new manual to handle it and the site personnel must be trained regularly. The operator manuals have been reviewed in the wake of Fukushima and further enhanced to cover the actual environment resulting from an earthquake and tsunami, additional accident management measures, preparation for total loss of electricity, etc. It is the responsibility of a plant operating organization to constantly continue the effort of safety improvement.

3.3. Natural circulation decay heat removal

Natural circulation decay heat removal was assessed in detail using the plant dynamics simulation computer code that was developed for safety evaluation of Monju and was validated through Monju and other plant data, including a natural convection test in the experimental fast reactor Joyo. In the event of SBO, the main cooling circuits are automatically switched to the air cooled auxiliary cooling systems in three loops that are bypassed from the secondary heat transport systems. Each auxiliary system is equipped with an air cooler that is placed on a higher elevation, such that the vertical height difference can promote the establishment of a natural circulation driving force. This is a passive safety mechanism and hence is generally more reliable than a safety system having active components such as pumps and blowers. It was concluded from our evaluation that decay heat removal by natural circulation from the core is possible with a sufficient safety margin as far as the coolant flow paths are available, and the reactor can be put into a cold shutdown state within three days. Actually, the establishment of natural circulation in one loop out of three is sufficient to eliminate excessive fuel heating.

The decay heat removal from the ex-vessel fuel storage tank, which is cooled by sodium in three loops, was also evaluated in detail. Although the electromagnetic pumps of the sodium loops are to be powered by a newly added power supply vehicle, even under SBO conditions, a natural circulation cooling feature is also available and the designed maximum decay power can be removed by two out of three loops. Spent fuel subassemblies were washed of sodium and 'canned' in containers which are finally stored in a fuel pool of water. The decay heat rate of the spent fuel cans when they are sent to the fuel pool becomes sufficiently low to eliminate the possibility of water boiling. The total heat generation balances the heat losses from the pool, and the water temperature stays below 70°C at most. Water evaporation from the surface is evaluated to be slow, such that it takes as long as three months before additional cooling water needs to be added.

3.4. Comprehensive safety assessment (stress tests)

Similar to other LWR plants in Japan, a comprehensive safety assessment has been conducted to evaluate the plant tolerances against extreme natural phenomena (earthquake, tsunami and a combination of the two) or the loss of important safety features (SBO, loss of ultimate heat sink and a combination of the two). This is the Japanese stress test and is applied to Monju as well.

Severe accident sequences that potentially lead to core damage are represented by event trees and the tolerances (or safety margins) of the safety (protection) systems or components under the beyond design basis conditions are evaluated one by one to determine which one is weakest. This is defined as a cliff edge, meaning that the core damage sequence results if this tolerance level is exceeded.

In the case of extreme earthquakes, for example, the weakest safety related component was evaluated to be a valve at the outlet sodium piping of the air cooler, which needs to be operated to establish a coolant path to the heat sink. The current design basis earthquake acceleration of Monju is 760 gal (0.78g), which itself has been increased since 2009 from the original design value (466 gal). The valve can withstand the acceleration level 1.86 times larger than the design basis. For tsunami, our design basis tsunami height is 5.2 m above sea level. Since the plant is built on a ground level 21 m above sea level, the tsunami design has a safety margin factor of 4.0. As described in the previous section, Monju has an advantageous safety feature for decay heat removal, with the air being the ultimate heat sink. It has been concluded that that there is no cliff edge effect under the conditions of SBO or loss of ultimate heat sink. The tolerances (safety margins) of the ex-vessel fuel storage tank and the fuel pool were also assessed.

4. ONGOING CHANGE OF NATIONAL NUCLEAR PROGRAMME

4.1. National debate on nuclear policy

Since the Fukushima accident, a future nuclear programme has been a matter of public debate in Japan. A series of public debates and hearings were held during the summer of 2012 with three nuclear options (0%, 15% and 20–25% of electricity by nuclear in 2030). The majority of the public seemed to support a nuclear phase-out scenario and this led to the report Innovative Energy and Environment Strategy issued by the Energy and Environment Council in September 2012.

Although the report aims at early realization of a non-nuclear society, some of the ongoing nuclear efforts are to be continued. For Monju, it is stated that a limited term research plan will be developed, implemented, reviewed and finished on compilation of achievement of R&D with Monju, and establishment of research programmes for the purpose of reducing the amount and toxicity of radioactive wastes.

4.2. MEXT working group on Monju research plan

On the basis of the above report of the Energy and Environment Council, a government level working group was formed in October in the Ministry of Education, Culture, Sports, Science and Technology (MEXT), a funding agency for the JAEA and Monju. An interim compilation of subjects of discussions and an overall direction of R&D policy with regard to Monju was drafted in December 2012.

The R&D programme under discussion is a limited scope, limited term plan, not simply to continue the original programme plan that existed before 11 March 2011, but to be reviewed from the beginning, assuming that a future plan to commercialize SFRs in Japan might no longer be given approval. In the working group, three R&D categories are being considered in conjunction with a careful prioritization argument:

(1) R&D with past achievements that are to be demonstrated through the SSTs and the following plant operation cycles. The five major sub-areas are: (i) core and fuel, (ii) components and systems design technology, (iii) sodium technology, (iv) plant operation and maintenance technology and (v) post-Fukushima severe accident technology. The priority argument includes those R&D items that can be performed only with Monju and key technology as a national SFR development programme.

(2) R&D aiming at technology development for reduction of radioactive waste volume and its radiotoxicity. The research includes burning (transmuting) long lived higher order plutonium isotopes and minor actinides. As already discussed in Section 2.2, the reactor core of Monju contains americium, as much as 1.5% of heavy metal. Thus, the operation of Monju itself means to demonstrate the safe and steady burning of americium-rich (and hence minor actinide-rich) MOX fuel. Taking advantage of international cooperation,

especially between France and the United States of America, a preparatory study is under way to plan an irradiation test programme using Monju, starting from a small scale pin level test.

(3) R&D for improving the safety of SFRs. Importance has been stressed since 11 March 2011 on prevention and mitigation of severe accidents. Although some severe accident sequences, such as unprotected loss of flow and transient overpower accidents, historically termed hypothetical core disruptive accidents, are treated during the licensing procedure, importance has been given to severe accident progression resulting from the SBO and loss of heat sink situations caused by tsunami. It is important to use Monju as a model plant to implement accident management guidelines and test both the hardware and software measures. Therefore, Monju can contribute to demonstration of post-Fukushima safety improvement measures.

Further discussions are to be made through the summer of 2013 and will include a knowledge management point of view, international cooperation, a management scheme of implementing R&D programmes, peer review and public comments.

4.3. Reform of Japanese nuclear regulation

In the wake of the Fukushima accident, national law and regulations to ensure the safety of nuclear installations have been amended significantly in Japan. This is an ongoing effort to be continued through 2013 and beyond. A new regulatory body, the Nuclear Regulation Authority, was established in September 2012. A new safety standard is being developed, taking into maximum account the lessons learned from the Fukushima accident.

For instance, the standard will explicitly call for severe accident management measures, redundant and more reliable power supply systems, reinforcement of decay heat removal capability and ultimate heat sink, as necessary. The enhanced design consideration of external hazards is another important issue. Although the safety standard under development is to be applied to LWRs, it is also to be considered in Monju, taking detailed consideration of the differences in safety features and design characteristics between LWRs and SFRs. The new regulation will also request the continuing effort of the licensees to further improve safety based on the licensee’s voluntary effort of safety consideration from a broad aspect, including probabilistic risk assessment.

5. PROSPECTS FOR THE FUTURE

Even in a wake of the Fukushima accident, we believe the roles of Monju as a prototype fast breeder reactor in Japan have not been changed. On the basis of the Joyo experience, Monju is expected to demonstrate, on a larger scale, that nuclear power generation is feasible and that the technology base has been made available. The data and experience in operation and maintenance of the plant are to be compiled and preserved for the design and operation of future plants. We believe it is extremely difficult to directly commercialize fast reactor technology without the technology and knowledge base of Monju and its operation and maintenance experience. The ongoing discussion in the MEXT working group seems to positively support this.

The future plan of SSTs and operation of Monju will be judged on the basis of a research plan developed by this working group through 2013. In parallel, we will keep up our efforts to improve the safety of Monju, applying the lessons learned from Fukushima. We believe the risk level of Monju will be made much lower with added accident management measures to further improve safety.

Even though the Fukushima accident and its radiological consequences have severely discredited the nuclear energy systems in Japan, the long term need for a stable energy supply and the global warming issue remain unchanged. SFR technology can provide a promising technology option to sustainably supply energy over centuries, and therefore the option will not be abandoned, especially in countries like Japan that have almost no domestic energy resources. The role of Monju as a prototype therefore continues to be important. In addition, it must be emphasized that some of the international joint research programmes using Monju are still actively continuing; the reactor is one of the very few fast reactor plants that are operable today. Thus, Monju is expected to play a role as an international asset to provide a research facility and knowledge/technology transfer to future generations.

6. CONCLUDING REMARKS

Monju, Japan's prototype SFR, restarted its test operation in 2010 after a long stoppage since the sodium leak accident in 1995. The zero power system startup tests were successfully conducted after long reactor shutdown. The major achievement was accurate prediction of reactor physics parameters with a core having a complex fuel composition that includes americium-rich fuel. The hardware troubles recently experienced, none of them safety significant, have been resolved to make the plant ready for the next power increase tests. The reactor, however, has been put into a standby mode again since the 11 March 2011

Fukushima-Daiichi accident, and a government level discussion is under way to determine the future direction of Monju with a prioritized research plan. We believe the roles of Monju will not change, i.e. further enhancing safety against severe accidents, demonstrating stable power generation and actinide burning, providing technology and a knowledge base for future SFRs and using the plant as an international research facility.

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INTERNATIONAL EXPERIENCE WITH FAST REACTOR OPERATION AND MAINTENANCE

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Abstract

This paper reviews the most important lessons learned from operation of the world's sodium cooled fast reactors, both test reactors and power producing reactors, which represent nearly 400 reactor-years of cumulative operating experience. The first reactor in the world to produce electricity was a fast reactor, the Experimental Breeder Reactor I, in December 1951. International experience with fast reactor technology exists in France, Germany, India, Japan, the Russian Federation, the United Kingdom and the United States of America. The operating experience with these reactors has been mixed; early problems were associated with fuel cladding, steam generators, fuel handling and sodium leakage. Excellent experience has been gained, however, that demonstrates the robust nature of the technology, the potential for exceedingly safe designs, ease of maintenance, ease of operation and the ability to effectively manage waste from spent fuel. It is a mature technology.

1. INTRODUCTION

As noted, the first reactor in the world to produce electricity was a fast reactor, the Experimental Breeder Reactor I, in December 1951. There were two principal reasons for this. First, it was relatively easy to design and build because the sodium-potassium coolant did not require a pressure vessel (liquid metal cooled systems operate at near atmospheric pressure). Second, the nuclear pioneers envisioned the need to fully utilize and extend the available fuel supply of uranium. Little was known at the time about the extent of uranium resources and a breeder reactor can multiply existing resources by a factor of as much as 100.

EBR-I was followed by EBR-II, which was a complete power plant. It was extremely successful, operating for 30 years and advancing the technology in many ways. Principal among its contributions were development of metal and oxide fast reactor fuel, operational safety tests which demonstrated the self-protecting nature of fast reactors, and fuel recycle technology that was efficient and secure. Perhaps the most important advance in safety was the demonstration of the self-protecting response of sodium cooled fast reactors in the event of anticipated transients without scram. Tests of loss of flow without scram and loss of heat sink without scram were conducted at EBR-II from full power with no resulting damage to fuel or systems, ushering in worldwide interest in passively safe reactor design.

Sodium cooled fast reactor operation has provided an extensive experience base. This experience base is fully supported by a combination of small test reactors that explored all aspects of the technology and larger operating reactors that provided power to the electric grid. Small experimental reactors were operated in the France (Rapsodie), Germany (KNK-II), India (FBTR), Japan (JOYO), the Russian Federation (BOR-60), the United Kingdom (DFR) and the United States of America (EBR-II). Power reactors and larger experimental reactors were operated in France (Phenix, Superphenix), Japan (Monju) the Russian Federation (BN-350, BN-600) and the USA (FERMI-1, FFTF). Current operating fast reactors are located in China (CEFR), India (FBTR) and the Russian Federation (BN-600, BOR-60). Fast reactors in Japan are currently shut down for repair and are awaiting restart authorization. These reactors are listed in Table 1 below.

TABLE 1. CHRONOLOGICAL OPERATION OF INTERNATIONAL FAST REACTORS

Reactor (country)	Thermal power (MW)	First criticality	Final shutdown	Operational period (years)
EBR-I (USA)	1.4	1951	1963	12
BR-5/BR1 (Russian Federation)	8	1958	2002	44
DFR (UK)	60	1959	1977	18
EBR-II (USA)	62.5	1964	1994	30
EFFBR (USA)	200	1963	1972	9
Rapsodie (France)	40	1967	1983	16

TABLE 1. CHRONOLOGICAL OPERATION OF INTERNATIONAL FAST REACTORS (cont.)

Reactor (country)	Thermal power (MW)	First criticality	Final shutdown	Operational period (years)
BOR-60 (Russian Federation)	55	1968		44
SEFOR (USA)	20	1969	1972	3
BN-350 (Kazakhstan)	750	1972	1999	27
Phenix (France)	563	1973	2009	36
PFR (UK)	650	1974	1994	20
JOYO (Japan)	50–100	1977		35
KNK-II (Germany)	58	1977	1991	14
FFTF (USA)	400	1980	1990	10
BN-600 (Russian Federation)	1470	1980		32
Superphenix (France)	3000	1985	1997	12
FBTR (India)	40	1985	27	
MONJU (Japan)	714	1994		
CEFR (China)	65	2010		2
PFBR (India)	1250	Under construction, 2013 estimated startup date		
BN-800 (Russian Federation)	2000	Under construction, 2014 estimated startup date		

The USA carried forward two separate tracks of technology development, primarily associated with the choice of fuel, metal or oxide, and plant configuration, pool versus loop. The first US commercial fast reactor, Fermi-I utilized metal fuel while the Fast-Flux-Test-Facility (FFTF) and the proposed Clinch River Breeder Reactor (CRBR) utilized oxide fuel. France, Germany, Japan and the Russian Federation all follow technology paths that use oxide fuel. It is worthwhile expanding this point because diversion of the technology paths has resulted in very different designs and performances, with the result that the EBR-II is somewhat unique in this family of reactors.

EBR-II was originally designed to demonstrate the technology introduced with EBR-I and to operate as a complete power plant along with its associated fuel cycle facility. Following this successful demonstration, its principal mission became to be an irradiation facility for the development and qualification of oxide fuel for the FFTF and CRBR. The fuel for these reactors was successfully developed (as demonstrated by the excellent 10-year operating record of the FFTF). When the FFTF assumed the major role in steady-state irradiation testing of oxide fuels, EBR-II was free to conduct more ambitious safety tests. For fuel, a major emphasis was to test the safety and operability of fuel with breached cladding, both metal and oxide. For plant safety, the emphasis was to demonstrate the self-protecting nature of fast reactors even when the reactor shutdown systems failed to operate. Owing to its unique contributions, emphasis will be given to lessons learned from the testing programme at EBR-II following a discussion of worldwide experience in areas of particular interest.

2. INTERNATIONAL EXPERIENCE

2.1. Physics

Fast reactor physics is unique in several respects. The neutron spectrum is hard (fast), the number of neutrons produced per fission is high, the core fissile content and enrichment are high, the core is not in its most reactive configuration and the neutron mean free path is long. These characteristics result in sensitive reactivity feedback to physical movement of reactor fuel and components, significant leakage of neutrons from the core, relative insensitivity to buildup of fission products and the ability to ‘fission’ minor actinides. These advantages and disadvantages have all been experienced in the international operation of fast reactors.

The sensitivity of cores to physical changes in core configuration has been seen in a number of ‘anomalous’ reactivity perturbations associated with reactor operation. At EBR-II, there was a problem encountered with ‘flowering’ of the core owing to problems of the core restraint design, subsequently solved with a redesign of the core and reflector assemblies. The Phenix reactor encountered anomalous reactivity perturbations in four instances, thought to be due to radial expansion of the core, possible ‘hanging’ of the core in this configuration followed by a rapid return to the original configuration. At the FBTR, anomalous reactivity perturbations were also experienced and thought to be associated with changes in core configuration associated with thermal gradients across the core. The lesson learned from these events is that proper design of fuel assemblies and core restraint systems is exceedingly important and all phases of operation must be considered. The result has been that no or few problems of like nature were experienced in subsequent reactors (FFTF for example operated without problems in this area). This is an area, like many others, where retention of knowledge is important to future designs.

The advantage of this sensitivity to physical changes is that very strong negative reactivity feedbacks may be induced on loss of flow events, as was demonstrated at EBR-II. It is a central feature of the self-protecting nature of sodium cooled fast reactors.

2.2. Sodium

Sodium as a coolant evokes many opinions. Some emphasize the negative, such as a strong chemical reaction with air or water at high temperature. Others praise the low operating pressure required of sodium cooling in fast reactors, its chemical compatibility with structural steel, ease of purification, its ability to chemically capture released fission products and ease of maintenance.

All operating fast reactors have experienced sodium leaks. One of the most famous occurred at the MONJU reactor in 1995. Approximately 640 kg of sodium leaked from the secondary sodium system resulting in a fire. Adverse publicity from the event resulted in the extended shutdown of MONJU and only recently has there been permission granted for a restart following redesign and modification of the thermowells where the leak occurred. A major sodium leak occurred at EBR-II in 1965 when a frozen sodium plug in sodium piping melted during maintenance, releasing approximately 100 kg of secondary sodium. As significant as these events were, no injuries resulted nor have any injuries apparently resulted from leaks at any other operating fast reactors. There are two reasons for this, the low pressure of the coolant which limits the rate of leakage and the improved understanding of designs to detect and prevent leakage. An important design feature for sodium piping is to limit penetrations and to conduct extensive vibration and failure analysis for penetrations into flowing sodium, such as thermowells. Suppression of the potential for fire can be achieved by surrounding the piping with guard piping if needed.

Sodium–steam interaction due to failure of steam generator tubing has also occurred, the most serious at the BN-350 reactor in its early operation. However, the BN-350 was not alone in this experience. All reactors using single wall evaporators have experienced steam–sodium leakage with the exception of Superphenix and the FBTR. In all cases, the failures were traced to poor welds, fabrication or design (EBR-II experienced no leaks with its duplex tube design). Much has been learned about prevention, detection and mitigation of the consequences of steam–sodium leakage but perhaps the most important is that such leaks are not catastrophic. As with other sodium leaks, no injuries have apparently resulted from the failure of steam generators. In addition, physical damage to the plants has been minor.

3. OPERATION AND MAINTENANCE

3.1. Reactor operation

It seems to be common wisdom that sodium cooled fast reactors are difficult to operate. Worldwide experience has demonstrated otherwise. At EBR-II, the reactor was operated with many different core configurations and at times with nearly 1/3 of the core made up of test assemblies with a variety of reactor fuels. In all configurations, the reactor was stable in its operation.

Another aspect is that operating procedures are straightforward, aided by the self-protecting nature of the reactors. At EBR-II, extensive tests were conducted that not only included anticipated transients without scram events associated with loss of flow, but also single rod run-out, primary pump control malfunctions, load following and steam system failures which demonstrated that operator action was not required to protect the reactor, even in the event of reactivity shutdown system failure. These tests also demonstrated that EBR-II was tolerant of errors of operator commission, namely taking an improper control action. These characteristics greatly reduced pressure on operators in the event of off-normal events. Rapid operator response was not required.

3.2. Fuel handling

Fuel handling in sodium coolant is challenging because there is no visual reference for the operations being conducted. This makes design of fuel handling systems especially important. Perhaps no fast reactor has had more fuel handling operations than EBR-II, >100 000 operations conducted successfully, but three errors occurred, any one of which could have terminated EBR-II operations. In each case dropped or damaged assemblies were successfully located and removed. Similar events have occurred at other fast reactors. The JOYO reactor suffered significant damage in 2000 when an experimental assembly was not successfully detached from the fuel handling equipment, resulting in damage to the upper internal structure. Significant repair will be required before JOYO can be restarted.

Lessons learned from this experience are that complete reliance cannot be placed on interlocks and instruments to ensure that fuel handling is progressing normally. At EBR-II, it was found that manual operation by operators at crucial steps in the fuel handling sequence was essential. Manual operations provide tactile verification that assemblies are properly engaged and limit the force that can be applied, so that damage will not occur if interferences are encountered. In conjunction with manual operation, acoustic monitors that alert the operator to physical contact (either normal or unexpected) have proven invaluable. There have also been significant efforts to develop under-sodium viewing technology, which certainly has promise. In addition, recent advances in visualization technology such as found in aviation (synthetic vision based upon known location and detailed mapping of terrain) has promise. The bottom line is that fuel handling systems deserve extraordinary attention in design, maintenance and operation.

3.3. Maintenance

Maintenance of sodium systems has been shown to be straightforward. Several aspects are worth noting. First, sodium systems operate at low pressure, which significantly reduces the hazards of leakage during maintenance. Second, since sodium is a solid when cooled below 98°C, it is possible to conduct maintenance on sodium systems without draining them. For example, if a valve needed to be replaced, the typical process is to freeze the sodium where the valve is located, cut the valve out of the system and replace it. These procedures have been well developed and demonstrated.

There are also many examples of successful removal, cleaning and repair of primary sodium system components such as pumps. At EBR-II, each of the two primary pumps was removed twice. The process is to draw the pump out of the primary sodium into an inerted container. Residual sodium is allowed to drain and the residue reacted with moist argon. Hands-on maintenance may then be conducted. At EBR-II, it was found that radiation exposure of maintenance personnel was very low, even though there were many maintenance procedures performed on sodium systems and components. Access to the reactor building was not restricted during reactor operation because radiation levels were always very low.

3.4. Safety

The safety record of fast reactors is outstanding. These reactors benefit from several operating characteristics that enhance safety. Perhaps the most important is that they operate at near atmospheric pressure. Also, the cores are relatively insensitive to spacial power shifts and fission product buildup such as xenon poisoning. Sodium is a very effective heat transfer medium, permitting high core power densities while effectively removing decay heat. Sodium is also non-corrosive to fuel cladding and structural components of the core and even when fuel cladding is breached, the fuel can be operated safely. Further, passive safety characteristics are easily achieved.

The technology is robust. Perhaps the best illustration of this is the Fermi-1 accident in 1966, which involved a partial fuel meltdown. The cause of the meltdown was a partial flow blockage from a loose plate below the core. The significance to safety of this event is that even though melting occurred in metal fuel assemblies directly impacted by the flow blockage, fuel failure did not progress across the core and no radioactive material was released from the reactor. Sodium combines with many of the chemically active fission products, such as iodine.

Much research was conducted in the 1960s, 1970s and 1980s to address the ‘hypothetical core disruption accident’, driven by a loss of flow event that resulted in a significant overpower transient associated with a positive sodium void reactivity coefficient and followed by core collapse and compaction. This research resulted in both a better understanding of the behaviour of fast reactor systems under these hypothetical scenarios and designs to ensure that such events could not happen. The most extensive physical research was carried out at the TREAT reactor in the USA which established that in the event of a major power excursion, core compaction would not occur but that fuel would disperse. At the other end of the spectrum, tests at EBR-II demonstrated that loss of flow without scram could be safely accommodated with inherent shutdown mechanisms and with no resulting damage. Positive sodium void reactivity coefficients can also be minimized through advances in core design.

3.5. Decommissioning

Decommissioning of several sodium cooled fast reactors has been successfully accomplished. The EBR-II reactor was successfully decommissioned after 30 years of operation and both the process and the technology were shown to be effective. Decommissioning EBR-II involved draining ~341 000 L (~90 000 US gallons) of sodium from the primary tank and reacting it with water to produce a 70wt% sodium hydroxide solution that is solid at room temperature.

Residual sodium in the systems was successfully removed by reacting it first with moist CO_2 followed by a water fill and ‘wash’. An interesting footnote to this process is that the primary tank and components contained therein was found to be ‘pristine’; there was no evidence of corrosion from the sodium coolant following 40 years of containing sodium at high temperature.

Perhaps one of the most impressive decommissioning efforts was that at BN-350, which went through a similar process with the exception that the sodium was highly contaminated with fission products (principally Cs-137) following years of operation with fuel with breached cladding. The sodium was cleaned before draining using a technique developed at EBR-II, flowing sodium through a specialized graphite/charcoal filter.

As with any reactor system, a key to successful decommissioning is to anticipate the need in the original design. For fast reactors, it is important that provisions for completely draining the sodium be provided, ensuring that there is no opportunity for residual ‘pools’ of sodium be trapped in spaces that will not drain.

4. EBR-II MISSIONS

4.1. Power plant operation

EBR-II operational capacity factors approached 80% even with an aggressive testing programme. Maintenance techniques were proven, with exposure to personnel less than 10% of that for a comparable light water cooled reactor. Effective sodium management was demonstrated, including successful suppression of a fire from a major sodium leak early in EBR-II’s operation and subsequent small leaks. The steam generators operated quite well, with no failures or leaks in the systems, a testament to the duplex tube design.

Fuel reprocessing was also very successful, with over 35 000 fuel pins reprocessed and recycled to the reactor in the first five years of operation. This demonstrated the viability of remote casting of metallic fuel elements and non-aqueous reprocessing of spent fuel using a simple melt refining process.

4.2. Fuel development

EBR-II metal driver fuel was significantly improved over the course of the 30 year operating life of the reactor. Burnups of in excess of 20at.% were achieved. A full range of metal fuel compositions was tested, including uranium-zirconium and uranium-zirconium-plutonium mixtures, with and without addition of minor actinides. Peak cladding temperatures reached 620°C with maximum in-reactor exposures of 5 years. An important conclusion is that metal fuel is a versatile and 'forgiving' fuel design, able to accommodate a wide range of compositions.

4.3. Operation safety testing

Operation safety testing involved integral plant safety tests as well as fuel safety tests.

Extensive tests, including both steady state and transient overpower conditions, demonstrated that metal fuel was completely compatible with the sodium coolant and a breach in cladding would not 'grow'. The safety case was made that breached cladding in metal fuel could be safely accommodated; no fuel loss would be expected.

The most dramatic of the safety tests were those involving the whole plant, leading to the inherent safety demonstration tests conducted in April 1986. The first of these was loss of all pumping power with failure to scram, simulating a station blackout with failure to scram. The reactor was brought to 100% power and the pumps were turned off, allowing them to coast down and coolant flow to transition from forced to natural convective flow.

Temperatures initially rose rapidly as the cooling flow decayed, but the increase in temperature introduced sufficient negative reactivity feedback that the power was also reduced rapidly, resulting in peak core coolant temperatures that were higher than for normal operation (~700^{o}C (~1300^{o}F) versus ~477^{o}C (890^{o}F) at normal operation) but not high enough to damage the fuel. A point to be emphasized is that there was no fuel or core damage with this event, unlike what would occur in a conventional reactor system. In fact, this was the 45th test of anticipated transients without scram events on this core and the reactor was restarted for a subsequent test that same afternoon.

In addition to the anticipated transients without scram tests, many tests were conducted to verify the response of the EBR-II plant to other off-normal events such as rapid movement of a single control rod, uncontrolled run-up of a primary pump and load following through changes in power demand at the steam turbine. These testing sequences were very successful and demonstrated the robust nature of the plant. Following the full complement of tests, a level 1 probabilistic risk assessment was completed to document the self-protecting nature of the reactor plant to off-normal events, even those as severe as without scram.

4.4. Prototype for the integral fast reactor

With all that was learned through EBR-II operation, the results were integrated into an approach to fast reactor design which was termed the integral fast reactor. A new feature of the approach was a reprocessing technology that accommodated fuel containing actinides and that offered proliferation resistance. The reason that the system offers proliferation resistance is that it is virtually impossible to separate Pu cleanly. Through a quirk of nature, the free energies of Pu and the minor actinides in the salt are so closely aligned that it is virtually impossible to adjust electro-refiner voltages to distinguish between them for transport of material.

An important aspect of the integral fast reactor fuel cycle was the production of waste forms suitable for geological storage; one was ceramic and the other metallic.

5. CONCLUSION

Worldwide experience with fast reactors has demonstrated the robustness of the technology and it stands ready for worldwide deployment. The lessons learned are many and there is danger that what has been learned will be forgotten given that in some countries there is little activity in fast reactor development at the present time. For this reason it is essential that knowledge of fast reactor technology be preserved, an activity supported in the USA as well as other countries.

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OPERATING EXPERIENCE WITH THE BN-600 SODIUM COOLED FAST REACTOR

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Abstract

The report considers the main design features of the BN-600 liquid metal fast reactor. The performance indicators achieved for 32 years of operation are given. The measures taken to enhance BN-600 reactor power unit safety and replace and extend its equipment lifetime and their results allowed the design lifetime of the power unit to be extended up to 40 years (until 31 March 2020) are presented. The considered integrated material, methodological and theoretical investigations justifying the serviceability of the irreplaceable components of the BN-600 reactor facility have shown that the strength conditions have not been violated in any of the critical reactor components after 45 years of operation. The results, both of the actions taken to enhance the BN-600 reactor power unit safety and corrective measures related to the events at the Fukushima nuclear power plant, allow the safety of the power unit exposed to any possible extreme external impact to be improved.

1. INTRODUCTION

On 8 April 2012, 32 years had elapsed since the BN-600 power unit was first connected to the power grid. As of 1 January 2013, the BN-600 reactor had accumulated 227.7 critical hours since the beginning of its operation, which corresponds to 26 years of reactor operation at rated power.

As a result of the operation of the BN-600 power unit, the design and actual performance indicators of the main equipment were shown to be in a good agreement. In this respect, the following high performance indicators were achieved:

— Gross efficiency of the unit: 42.6 %.
— Capacity factor for the entire period of operation: 72.6% (74.79% excluding the commissioning period).

For 2012, the capacity factor was 81.15%. Figure 1 shows a variation of the capacity factor for the period of commercial operation of the Beloyarsk nuclear power plant, power unit No. 3.

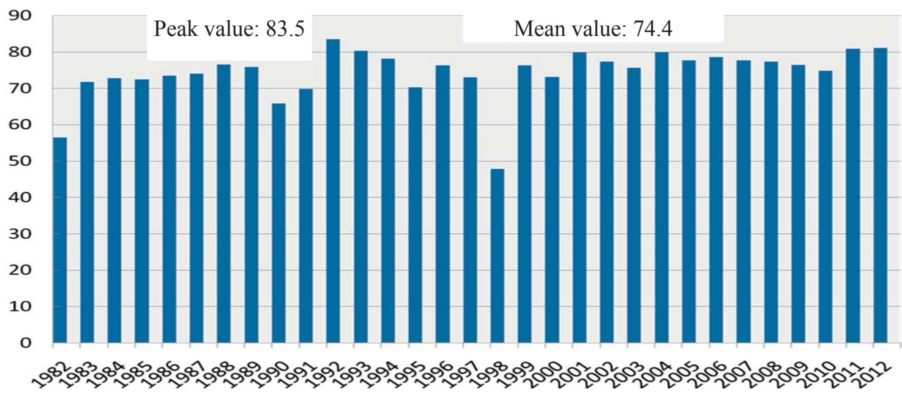

FIG. 1. Capacity factor variation for the Beloyarsk nuclear power plant, power unit No. 3, commercial operation.

After 32 years of operation, only six values of the capacity factor are less than 70%, of which three fall in the initial period of operation and are associated with the mastering of the operating modes, both of the power unit as a whole and its individual components [1–4]. These three values were for:

(i) 1990 (the capacity factor of 65.9%): the losses of electrical generation were mainly associated with a leak in the cooling system of the generators.
(ii) 1991 (the capacity factor of 69.8%): the failures of the water–steam circuit equipment and one sodium–water reaction occurred in the steam generator.
(iii) 1998 (the capacity factor of 47.9%): the planned repairs of the central rotating column of the reactor.

The following main features that distinguish the BN-600 reactor design from the designs of the previous reactor facilities should be mentioned:

— Pool layout of the primary circuit;
— Modular staged design of the steam generator.

The period of operation of BN-600 can be conventionally divided into two intervals, i.e. from April 1980 to September 1981, the period of bringing it from 30% to 80% of rated power in successive steps, and from October 1981 up to now, the period of commercial electrical generation.

In the second interval of BN-600 operation, a capacity factor as high as 74.4% was achieved. The achieved annual capacity factor is the maximum with the present length of the planned outages of the power unit. During recent years, the power unit has been steadily operating with an electrical load of 600–620 MW.

At the present time, the duration of the planned inspections of equipment is mainly determined by the standard duration of the overhaul of the turbine sets and the need for refuelling the reactor twice a year. The unplanned losses account for 2.2% of the time. Table 1 shows the main BN-600 performance indicators achieved both for a 32-year period and for 2012.

TABLE 1. MAIN BN-600 PERFORMANCE INDICATORS

Indicators	Measurement units	For 2012	For operation period	The best value
Electric power	MW	610	600	625[a]
Running hours	hour	7376	233 631	7449
Number of unplanned shutdowns	pieces	N/A	23	0
Electrical generation	million kW·h	4130.6	124 988.07	4401.96
Capacity factor	%	81.15	74.4	83.52
Annual collective exposure dose	man·Sv	0.42	20.86	0.08
Radioactive noble gas emission	Ci/year	103.2	40 581	60
Efficiency	%	42.60	40.8 (design value)	

[a] Peak achieved value.

2. OPERATING EXPERIENCE FROM BELOYARSK NUCLEAR POWER PLANT POWER UNIT No. 3

2.1. General information

During operation, 63 reactor core refuellings have been performed and along with this, a maximum fuel burnup of 11.1% of h.a. for standard fuel subassemblies and 12.0% of h.a. for experimental (non-standard) subassemblies were achieved.

During the operation, both a high degree of radiation safety of the BN-600 power unit with its long term operation at the rated power parameters and a minimal impact on the environment were ensured. The results of measurements of the radiation background and activity of process media show that their values do not exceed the design ones. The radiation levels in the attended and semi-attended premises of the reactor building do not exceed the regulatory values, while the levels of the gamma radiation on the secondary circuit equipment are within background values.

The gas–aerosol emissions into the environment are well within the accepted limits and since 1988 have been 2 Ci/d or less; when the unit operates there are actually no radioactive liquid discharges. The average amount of solid radioactive waste is 22 m^3/year. Since 1984, unit personnel exposure has been at the level of 30–60 man·cSv.

During the BN-600 operation (as of the end of December 2011), there were a total of 117 violations of normal operation that led to the unplanned decrease of reactor power. Figure 2 shows the distribution of these violations by year.

As can be seen from Fig. 2, most failures with reduced power occurred at the initial stage of operation of the BN-600 power unit, i.e. during the mastering of this reactor technology by personnel.

In recent years, the decrease in the capacity factor due to unplanned shutdowns and reductions in reactor power have been accounted for by the failures of the conventional power generating and power equipment related to the water–steam circuit, i.e. by failures not directly connected with fast reactor technology features.

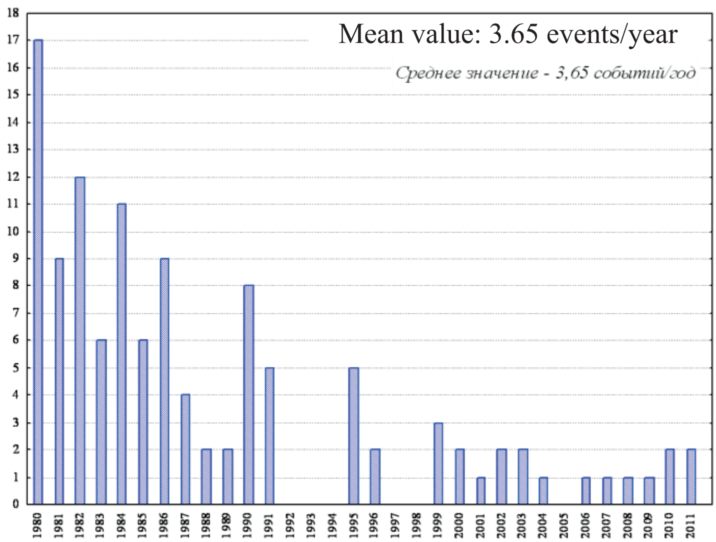

FIG. 2. Number of violations which led to reduction in power.

It should be noted that no violation of the normal operation of the power unit resulted in the exposure of the population or the environment to radiation, and by the ‘off-site impact’ parameter all the violations were classified as non-essential.

The operating experience from the power unit and the tests performed on it mainly validated the adopted design solutions, but at the same time they identified a number of shortcomings in some equipment that caused the need for their upgrade and backfitting. Therefore, in the organizations that accompanied the operation of the BN-600 power unit and at the power unit itself, an extensive programme of R&D was undertaken, which resulted in some components and systems being upgraded and the operating modes of the power unit improved, in particular, the reactor core, the electric drives of the primary and secondary sodium pumps and the equipment of the reactor refuelling systems were upgraded, new failed fuel detection systems were created, new systems of reactor vessel integrity monitoring and new systems of monitoring the displacement of the reactor vessel and the auxiliary primary sodium pipelines were developed, the steam generator’s water–sodium reaction detection systems and the systems for the technical diagnostics of the reactor and steam generator equipment were considerably improved, the lifetime extension of some equipment was justified and much more.

2.2. Core

During 1986–1987, the first modification of the core was carried out with the introduction of three fuel enrichment zones, i.e. a low enriched zone (17%), an intermediate enriched zone (21%) and a high enriched zone (26%), and the height of the fissile part was increased from 750 mm to 1000 mm. The peak fuel burnups were 6.5% of h.a. in the low enriched zone subassemblies, 6.9% of h.a. in the intermediate enriched zone subassemblies and 8.3% of h.a in the high enriched zone subassemblies. The changeover to modified core 01M allowed the length of the power unit cycle to be increased from 100 to 165 days.

In 1991–1993, the reactor was changed over to the second modification core (core 01M1) with a peak fuel burnup of 10% of h.a. To ensure a higher fuel burnup in the second modification core, the fuel loading in the subassemblies was increased at the expense of the increase both in the height of the fissile part from 1000 mm to 1030 mm and in the effective density of the fuel in the fuel pins, from 8.5 g/cm^3 to 8.6 g/cm^3. The length of the power unit cycle for the modified core 01M1 was 160 days.

Since 2005, the BN-600 reactor has been changed over to core 01M2 with a peak fuel burnup of 11.1% of h.a. At the BN-600 reactor, the subassemblies with the fuel pins both with the pelletized and vibrocompacted MOX fuel are tested.

As result of the modification of the core, including for the purposes of preventing fuel failures, the following took place:

— The peak linear rating was decreased from 540 down to 480 W/cm at the expense of the increase in the core height from 750 mm to 1000 mm.
— The new structural materials for the subassembly wrapper (cold worked ferritic-martensitic steel EP-450) and fuel cladding (cold worked ChS-68) were utilized.

2.3. Main sodium equipment

In the initial period of operation, a number of the problems associated with the primary and secondary sodium pump failures due to their excessive vibration, shaft cracking, damage to the half-clutches and unreliable operation of the electric drive were successfully solved. A vibration monitoring system of the primary and secondary sodium pumps was introduced. The strain measurement of the shafts allowed the presence, both of the resonances of the shaft's torsional oscillations in the operating range of the rotation frequencies and coincidence of the pulses of the output power of the electric drive with the natural frequency of the torsional oscillations of the shaft, to be found.

By the results of the examination, the primary and secondary sodium pump shafts were replaced with the upgraded ones, the clutches engaging the pump shaft with the motor rotor were redesigned and the primary and secondary sodium pump motors were changed over to the uncontrolled mode of the short circuited rotor with the power unit operating at rated power. In addition, the gastight shaft seal, the brush-and-contact assembly and the attachment point of the tachogenerator of the electric drive were upgraded. The measures taken allowed the failures of the primary and secondary sodium pumps resulting from the above-mentioned causes to be eliminated and reliability of the primary sodium pumps to be essentially improved, as well as for the primary sodium pump lifetime to be increased from 20 000 to 57 000 hours and for the secondary sodium pump lifetime to be increased from 50 000 to 125 000 hours.

The lifetime of 30 years was built into the strength calculations of the design of the intermediate heat exchanger, taking into account the absence of operating experience with similar equipment and the design organization having limited the lifetime of the heat exchangers to 20 years. On the basis of the calculations, based on the modern standards and taking into account the results of the positive operation as well as the results of the metal examination and inspection, the set lifetime of the intermediate heat exchangers was successively extended to 22 and then 25 years. In 2005, the decision to extend the set lifetime of the intermediate heat exchangers up to 30 years was arranged.

In the period 25 March to 26 April 2006, intermediate heat exchanger 5A was replaced. The examination of the intermediate heat exchanger metal confirmed the serviceability of the intermediate heat exchanger up to 2010 and justified the feasibility of operation up to 2025. The lifetime was increased 2.2 times, from 20 to 45 years.

The operating experience from the power unit showed that the utilized modular staged steam generators have a high operational reliability. Taking into account the structural materials used and the built-in operating modes, the lifetime for the evaporator stages of the PGN-200M steam generator was set to be 50 000 hours and the lifetime for other steam generator components, including the superheater and reheater stages, was 200 000 hours.

The operation of the steam generator at rated power with seven of eight modules belonging to the steam generator of an individual loop was validated to be feasible by experiment. As a result, when the failures with sodium–water reactions occurred in the PGN-200M stages, the loop needed to be disconnected only twice and the unit needed to be shut down only once. The examination of the damaged stages showed that the sodium–water reactions were most likely caused by manufacturing flaws that were undetected using the standard inspection techniques during the manufacturer’s tests.

Over the period of operation of the BN-600 Unit 12, sodium–water reactions at the steam generator stages were registered, i.e. one at the evaporator stage, five at the superheater stages and six at the reheater stages.

Over the period of operation, a number of the following jobs aimed at improving of the reliability of the steam generators were fulfilled:

— Optimization of the prestart and reagent cleanings;
— Elimination of the design deficiencies of the main sodium valves;
— Upgrade of the seal assemblies of the covers of the reheater stages;
— Backfitting of the auxiliary pipelines;
— Upgrade of the sodium–water reaction detection systems.

The in-service justification of the evaporator lifetime of up to 105 000 hours (instead of the 50 000 hours envisaged by the designer) made it possible to change over to one-time replacement of the evaporators instead of the planned three-time replacement during the life of the power unit.

To date, the evaporator lifetime has been extended up to 125 000 hours.

In the framework of the work conducted on power unit No. 3 lifetime extension, all 72 stages on all three steam generators were replaced.

2.4. Reactor components and mechanisms, including the refuelling system

The design features of the BN-600 reactor do not allow the internals to be repaired or replaced. Taking this circumstance into account, the designer of the reactor facility, including the reactor vessel and internals, justified their lifetime of 30 years by calculation.

To date, with the reactor facility chief designer involved, the reactor has been justified to be serviceable beyond the design lifetime, for up to 45 years of operation on the basis of the integrated material, and methodological and theoretical studies of the irreplaceable components of the BN-600 reactor facility.

The spent subassemblies are unloaded and the fresh subassemblies loaded on to their places in the completely closed environment. In 2010, the ‘SUPER-505’ system was replaced with the digital control system of the fuel handling system (‘refuelling’ and ‘cleaning’ subsystems). It was manufactured by the scientific production association “Avtomatika”.

In 1998, work on the elimination of the reactor central rotating column jamming which had been experienced since 1995 was successfully completed and the phenomenon of jamming of the central rotating column was corrected. With regard to fast reactors, this work was carried out for the first time in the world.

2.5. Operating experience from the sodium circuits

The BN-600 reactor was designed following the principle of the pool layout with all the primary components being enclosed within the same tank (vessel). The vessel has no branch lines below the level of sodium. To prevent leakage of radioactive sodium, all the pipelines of the auxiliary systems outside the reactor vessel are enclosed in the guard casing up to the gate valves, including the bodies of the valves themselves.

The external primary sodium systems are functionally divided into a number of the following systems:

- Flowmeter loop;
- Primary sodium purification system;
- Primary circuit tanks with the reactor overflow and gas balancing system;
- Loop for spectrometry and detection of the fission products in the primary sodium.

The isolating valves on the primary sodium system pipelines are equipped with electric drives and when the reactor is operating at power operation these valves are open. In the case of the loss of integrity of the pipelines and equipment without guard casings, the sodium leak is detected by the following:

- The ground short circuit of the helixes of the heat zones of the trace heating of the affected section;
- Use of the fire annunciators installed in this room;
- Use of the air radiation and temperature monitoring system in this room.

The affected section is isolated and drained of sodium.

Since the beginning of the reactor operation, the only sodium leak necessitating power unit shutdown occurred in the auxiliary primary circuit system.

As a result of this sodium leak, the individual rooms of the reactor building were contaminated with radioactive Na-24 (half-life of 15 h). There was no contamination of the territory within and outside the site.

The main feature of the sodium cooled reactor layout is the presence of the intermediate (secondary) sodium circuit. The secondary circuit has a number of the following functions:

- The prevention and retention of radioactivity in the primary circuit during the incidents with the loss of integrity of the heat transfer surfaces of the heat exchangers;
- The heat transfer from the primary circuit to the steam generator;
- The prevention of ingress of water/steam into the primary circuit.

Unlike the primary sodium, the secondary sodium is not radioactive and in the case of its leakage, any environmental contamination is ruled out. The main hazard of the secondary sodium leakage is connected to its combustion in air.

Over the entire period of operation of the BN-600 reactor, four incidents occurred with secondary sodium leaks which led to the total loss of capacity factor of 0.13%. Two sodium leaks occurred on the drainage lines of the steam generator stages. The first one was minor and was due to a flaw in a welded joint, while the second one was caused by a flaw in the parent metal.

As a drawback of the modular staged steam generator, the need to install a significant number of secondary sodium gate valves designed to disconnect the failed steam generator modules can be mentioned. At the BN-600, for these purposes, the wedge gate valves with an internal diameter of 300 mm with the freezing stem packing and sealing of the gate valve cover to the body by welding are used. During operation, the sodium leaks in this welded joint occurred repeatedly. In two cases, a heat transfer loop required to be disconnected because the valves were located on the non-isolatable sections of the pipelines.

The gate valve leakages occurred due to technological defects which arose due to the shortcomings of the technological process of repair and testing of the welded joints. The technological process was modified and additional detectors were installed for early sodium leakage detection. The subsequent period of operation confirmed the effectiveness of the measures taken and no sodium leaks on the gate valves across this joint occurred.

2.6. Sodium technology

During the construction and commissioning of the BN-600 reactor, much of the experience gained was in the field of the technologies related to the production of reactor purity sodium, its delivery to the plant and the filling of the sodium circuits of the power unit. Effective systems for the detection of various impurities in sodium and its cleaning of them using cold traps were also developed.

To date, the cold trap regeneration technique developed at the Institute of Power Physics and Engineering has been mastered at the Beloyarsk nuclear power plant. The recovery of two cold traps of sodium impurities has already been successfully completed. Now, work is in hand to create the standard regeneration system.

It should be noted that the power unit personnel thoroughly mastered the operation of the sodium circuits with large amounts of sodium, including the technologies for inspection, repair and replacement of the sodium equipment and pipelines, which allowed sodium leakages to be virtually eliminated.

2.7. Main power equipment and water–steam circuit equipment

As a whole, the turbine generators and water–steam circuit heat removal system are characterized by successful operation. Since the beginning of operation, the main pipelines have had no serious incidents.

The electric generators are conventional and tried-and-tested equipment. However, during the entire period of their operation, there were cases of leakage of the stator cooling water system. On several occasions this led to the unplanned trips of the generators. The reason was a fault of the designer of the stator bar seals.

2.8. Improvement of the power unit operation modes

Taking into account the accumulated experience gained, a number of the following BN-600 power unit process modes of operation were corrected and improved:

— The mode of connecting the heat transfer loop with the power unit in operation was improved;

- The operation of the steam generator without one of eight modules was justified;
- The mode of connecting a steam generator module with the power unit in operation was developed and brought into operation;
- The mode of the reactor shutdown with the actuation of the slow emergency protection was ruled out;
- The reactor decay heat removal mode was improved;
- The power unit startup mode was improved.

3. LIFETIME EXTENSION

The BN-600 belongs to the second generation of nuclear power units and its 30-year design lifetime expired in April 2010. After a positive prediction of the residual lifetime of the main components of the power unit had been obtained and results of the assessment of the economic expediency had been made available, it was decided to extend the BN-600 lifetime.

The work on the integrated examination of the power unit and evaluation, both of its safety and serviceability of the irreplaceable reactor components and structures, as carried out in 2003–2005, showed the feasibility and economic expediency of extending the BN-600 lifetime by another 15 years.

Since 2005, the work to upgrade and replace the equipment has been carried out at the power unit. The work was completed in two fields, i.e. on improving safety and eliminating derogations of the regulations as well as on replacing the worn-out equipment. In the first field the following were foreseen:

- Construction of the standby control desk;
- Creation of the second set of emergency protection equipment;
- Installation of the additional emergency cooling system with the ‘sodium-to-air’ heat exchangers;
- Improvement of the seismic stability of buildings, structures and equipment;
- Improvement of the reliability of the emergency power supply system;
- Upgrade of the radiation monitoring system.

In the second field the following components were replaced:

- Steam generators (72 stages);
- Impellers of the primary sodium pumps;
- Feedwater pumps;

- Blades, wheels and diaphragms of the last stages of the turbine low pressure cylinders;
- Turbogenerator excitation system, etc. (in total, the components of 19 systems).

The unique character of the work completed within the scope of the BN-600 lifetime extension is that it has taken place during the planned outages of the power unit, i.e. the BN-600 has continued to operate in the standard mode. After the major work had been completed, the licence to operate the BN-600 until 2020 was obtained.

As a new lifetime extending to 2025 for the components of the power unit has been formally established, it is possible to further extend the licence by another 5 years.

4. ACTIONS TO ENHANCE SAFETY AND PREVENT AND IMPROVE THE CONDITIONS OF MANAGEMENT OF BEYOND DESIGN BASIS ACCIDENTS

After certain events at the Japanese nuclear power plants, an additional analysis of the resistance of power unit No. 3 to the impact of external factors (earthquake, wind, inundation, snowfall, high and low air temperatures, external fires, etc.) using the methodology proposed by the Western and Russian regulatory authorities was conducted.

Using the results of the investigations, confirmed by the Russian Technical Supervisory Authority's evaluation, the following could be mentioned:

- Bearing in mind the measures taken at the power unit, its resistance to the site specific external impacts has been confirmed.
- There is no need to expand the designer's list of the beyond design basis accidents.

At the same time, to improve the power unit's safety, in the case of extreme external impacts, the following were done:

(1) The main components ensuring the possibility of reactor heat removal, i.e. main building, reactor, primary and secondary circuit equipment of the reactor facility, including pipelines and steam generators, and a complex of the reactor refuelling mechanisms were investigated and confirmed to be stable and able to maintain the safety functions during an earthquake of magnitude 7 (i.e. 1 point above the safe shutdown earthquake).

(2) The power unit was additionally equipped with the following:
   (a) The 0.4/6 kV, 2 MW mobile diesel generator set for power supply of the most critical mechanisms;
   (b) The 0.4 kV, 0.2 MW mobile diesel generator set for power supply of the instrumentation and standby control desk as well as for boost charge;
   (c) The mobile pumping unit PNU 250/150 for the water–steam circuit replenishment;
   (d) The mobile pumping units PNU 150/120 and 500/50 to be used in the additional system for maintaining the water level in the spent fuel cooling ponds.

(3) The working documentation on the following was developed:
   (a) Both the additional system for the return of leaks from the spent fuel cooling pond and the system for replenishment of the spent fuel pond with water from the truck tank;
   (b) The additional insulation of the wall constructions of the buildings for conditions of extreme cold (minus 61°C) ambient air temperature;
   (c) The reinforcement of the wall panels and some load-bearing structures and replacement of the daylights for the extreme climatic loads;
   (d) The reinforcement of sections of the main pressure circulating water pipelines.

5. CONCLUSION

The operating experience from the BN-600 reactor power unit for more than 32 years is positive in terms of the demonstration of the feasibility of utilization of a sodium cooled fast reactor for commercial electricity generation [5].

The BN-600 reactor is an important key link, ensuring the continuity and succession of the development of the fast reactors in the Russian Federation, of which their reliable and steady operation confirms the good prospects for this line of the nuclear power industry.

During the course of the BN-600 power unit operation, valuable operating experience was gained from the individual systems and components, which should be preserved and utilized when developing the advanced designs of sodium cooled fast reactors.

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TWENTY-SIX YEARS OF OPERATING EXPERIENCE WITH THE FBTR AND FEEDBACK FOR FUTURE REACTOR DESIGN

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Abstract

India has limited uranium, but abundant thorium resources. For better utilization of uranium and to use the available thorium, a fast reactor programme is indispensable for India because fast reactors can generate electricity and breed additional fissile materials for future reactors. The Fast Breeder Test Reactor (FBTR) has provided a valuable test bed for the performance assessment of unique carbide fuel, materials, etc., experience in safe handling of sodium, in addition to generating employment of human resources. The knowledge gained through successful operation of the FBTR for the past 26 years has provided vital inputs for the commercialization of the fast breeder reactor programme through the construction of the Prototype Fast Breeder Reactor (PFBR). The PFBR is a 500 MW(e), sodium cooled, pool type fast breeder reactor currently under construction. It is essential to reduce the capital cost of future fast breeder reactors to make them competitive with thermal reactors. Operating experience gained with the FBTR provides vital input towards simplification of the design, improving its reliability, enhancing safety and achieving overall cost reduction. This paper includes a summary of 26 years of operating experience gained with the FBTR and its feedback into the PFBR design.

1. INTRODUCTION

The Fast Breeder Test Reactor (FBTR) at Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam, is a 40 MW(th) (13.6 MW(e)) sodium cooled, loop type, plutonium rich mixed carbide fuelled fast reactor which serves as an irradiation facility for development of fuel and structural material for future fast reactors. It has two primary and secondary loops and a common steam–water circuit with once-through steam generators (SGs), which supply superheated steam to the condensing turbine. There are two SGs per loop and these are located in a common casing. The SGs are not insulated in order to facilitate decay heat removal through the casing by natural convection to air. A 100% steam dump facility is provided in the steam–water circuit so as to allow the reactor to operate at full power for experimental purposes, even when a turbine is not available. The basic conceptual design of block pile, primary loop and reactor instrumentation are similar to the French reactor Rapsodie, whereas the steam–water circuit and turbogenerator are of indigenous design (Fig. 1).

The reactor power control, as well as shutdown, is achieved by six boron carbide control rods (90% enriched in B-10), which are held by the control rod drive mechanism. The control rod drive mechanism is supported at the top and is free to expand axially downwards. The fuel subassemblies are held at the bottom in the grid plate and are free to expand axially upwards. Being a small fast reactor with all feedback coefficients negative, only one type of shutdown system is provided. Also, two control rods are sufficient to bring the reactor to a cold shutdown state from full power.

The reactor attained its first criticality in October 1985, with the Mark I core consisting of 22 fuel subassemblies of 70% PuC + 30% UC fuel. In May 1987, while carrying out a low power physics experiment (<500 kW·t), a fuel handling incident took place and the reactor could resume operation only in May 1989 after recovering from the incident. Subsequently, low power physics and engineering experiments up to 1 MW·t were completed in 1992. In December 1993, after completion of commissioning of the SG and its leak detection system, reactor power was raised to 10.2 MW·t for the first time. Before resorting to steady power operation, it is mandatory to carry out important safety related high power engineering tests to validate all assumptions made in the safety evaluation report to ensure plant safety under various anticipated incidental situations. The data obtained from these tests also helps the designer to improve mathematical modelling for better prediction. Since these tests required special/off-normal plant conditions, detailed procedures were prepared and prior clearance obtained from safety authorities. Accordingly, a series of safety related engineering tests were conducted in 1992–1993, simulating the various incidents postulated in the safety

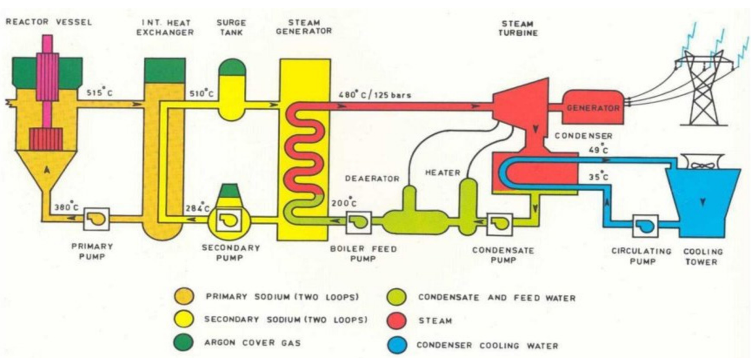

FIG. 1. Simplified schematic of the FBTR.

analysis. These included natural convection tests in the primary and secondary circuits, with the reactor operating at low power, simulating the decay heat. The results of these tests are very useful in gaining confidence in the capability of the decay heat removal systems and analysing various transients in the PFBR systems. After completing high power engineering and physics tests, reactor operation at high power was continued. Commissioning of the turbogenerator and its auxiliaries was subsequently completed and the turbogenerator was synchronized to the grid, producing 1.2 MW(e) in July 1997.

The reactor core was gradually enhanced by adding Mark II (55% PuC + 45% UC) and MOX subassemblies (44% PuO_2 + 56% UO_2) and power was raised in steps to 20 MW·t. Experiments conducted showed that MOX fuel of this composition is compatible with sodium. The fuel pins of the Mark I and Mark II compositions were irradiated in the reactor and discharged for post-irradiation examination to assess the fuel performance. The reactor was operated at 8 MW·t for irradiation of the zirconium–niobium (Zr-Nb) alloy for the PHWR programme. Nineteen irradiation campaigns have been completed so far. The MOX fuel (29% PuO_2 + 71% UO_2) uranium was enriched with U-233 to achieve the design linear heat rating of 450 W/cm required for the Prototype Fast Breeder Reactor (PFBR) and was irradiated to 112 000 MW·d/t to study its behaviour. Post-irradiation examination of the PFBR test fuel has been completed. More than 1000 pins of Mark I subassemby composition were irradiated up to a peak burnup of 155 000 MW·d/t and 61 pins of a lead subassembly were irradiated up to a peak burnup of 165 000 MW·d/t so far. Towards designing and building future metallic fuelled reactors, irradiation of metallic fuel pins has been commenced from the 18th campaign onwards. The reactor parameters achieved so far are given below.

Power	20.3 MW·t
Linear heat rating	400 W/cm for Mark I subassembly and 450 W/cm for PFBR test subassembly
Peak burnup	165 000 MW·d/t for Mark I subassembly 112 000 MW·d/t for PFBR test subassembly
Total operating time	45 368.7 h
Total thermal energy developed	413 950.51 MW·h
Primary sodium flow	800 m^3/h
Reactor inlet/outlet temperatures	390/490°C
Feedwater flow	33 T/h
Feedwater temperature	195°C
Steam conditions	470°C at 120 kg/cm^2

Reactor physics experiments were continued from initial criticality to full power operation. Initial core loading and approach to criticality was followed by several physics tests, such as measurement of control rod worth, subassembly worth, feedback reactivity coefficient related to power, coolant void, coolant temperature, coolant flow and cover gas pressure, kinetic experiments, flux tilting and flux measurement experiments. These tests were conducted to ensure the inherent safety of the reactor and to validate the assumptions made in the computation of safety analysis.

2. OPERATING EXPERIENCE FEEDBACK FROM THE FBTR [1–7]

2.1. Reactor operation

The FBTR was operated at various power levels up to 20.3 MW·t to date. During steady state operation, no difficulty has been experienced in maintaining reactor power within a small band. The FBTR experience indicated that reactor power could be controlled manually without any difficulty. The PFBR, being a power reactor, is expected to be operated at a steady power level most of the time. Sufficient confidence has been obtained for the smooth manual power control in the FBTR and hence for the PFBR also.

2.1.1. Optimization of reactor trip parameters

On the basis of operating experience, reactor trip parameters were optimized by modifications/additions/deletions of parameters in scram and the LOR circuit to avoid spurious trips and achieve maximum availability without compromising the safety of the reactor. Optimization of trip parameters was carried out for the PFBR and only safety action scram was provided. For example, scram is initiated in the case of high inlet temperature to the core to ‘sense’ flow reduction in the heat transport circuits, thus eliminating the need for LOR.

2.2. Fuel

The reactor was initially loaded with a small Mark I fuel core rated for 10.5 MW·t at a linear heat rating (LHR) of 250 W/cm. Being an untested fuel, the target burnup was initially set at 25 000 MW·d/t. The LHR and target burnup values have been progressively enhanced to 400 W/cm and 155 000 MW·d/t based on post-irradiation examination of Mark I fuel at different burnup levels [1]. At each stage of the LHR and burnup enhancement, rigorous theoretical analysis was carried out and safety clearances were obtained (Fig. 2).

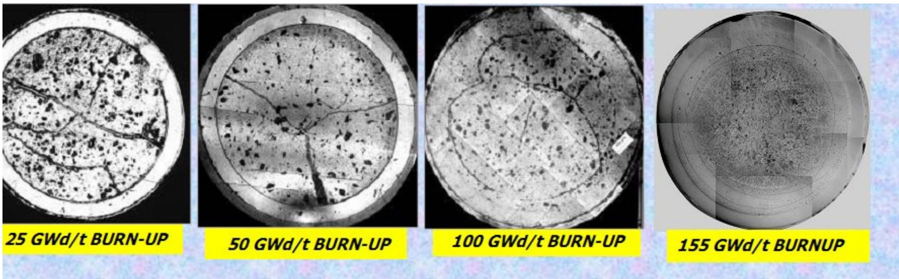

FIG. 2. Micrographs of Mark I fuel at different burnup levels.

The lead Mark I fuel (70% PuC + 30% UC) subassembly was irradiated up to a burnup of 165 000 MW·d/t without fuel clad failure. The fuel has also been successfully reprocessed at IGCAR. The successful closure of the fuel cycle was also demonstrated recently when a fuel subassembly with Pu recovered from FBTR fuel was loaded back into the core of the FBTR.

2.2.1. PFBR test fuel pin irradation

As the fuels used in the FBTR and PFBR are of different compositions, the PFBR test fuel was irradiated to 112 000 MW·d/t at the rated LHR of 450W/cm in the FBTR and post-irradiation examination of this subassembly was also completed. This is one of the major missions accomplished by the FBTR.

2.2.2. PFBR test fuel pin irradiation for initial gap closure study

During the 15th irradiation campaign, one experimental fuel pin with MOX fuel pellets of PFBR composition was irradiated in FBTR for 13 days at a linear power of 400 W/cm to gain an understanding of the beginning of life gap closure behaviour.

The fresh fuel is to be operated at a lower LHR until the gap closes by cracking and restructuring of the fuel. The duration of operation at a low LHR has implications on the economics of power generation. This experiment was aimed at optimizing the duration of pre-conditioning of the MOX fuel in the PFBR at lower linear power (400 W/cm) before raising the linear power to the design value of 450 W/cm. During post-irradiation examination (Fig. 3), the measurements at different locations indicated that the apparent gap had reduced from the average pre-irradiation value of 75–110 microns to a uniform value of around 20 microns in all the locations. The gap reduction during the beginning of life indicates the feasibility of increasing the LHR of PFBR fuel to the design value of 450 W/cm after the initial pre-conditioning of approximately 20 EFPD.

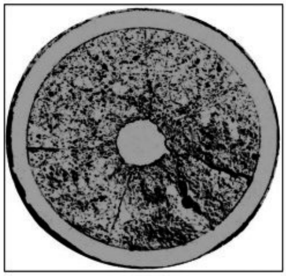

FIG. 3. Photo mosaic of PFBR test fuel pin.

2.2.3. Fuel subassembly clad failure incident

During the 17th irradiation campaign at 18 MW·t, one Mark I subassembly which has reached a burnup of 148 MW·d/t failed. The DND contrast ratio between west and east side during the reactor scram was observed to be more than 4.5 and the predicted burnup of the failed fuel subassembly based on the observed Kr^{85}/Kr^{88}activity ratio was more than 100 GW·d/t. Hence, from the above, it was inferred that any one of the highly burnt fuel subassemblies located in the west side of the 3rd ring is the failed subassembly. By the neutron flux tilting method, the failed fuel was identified and further confirmed by operating the reactor at high power after shifting it to the storage location. The subassembly was discharged from the reactor after its decay heat came down to acceptable limits for dry storage. In the PFBR, the DND blocks are provided at the inlet of the IHX in the hot pool. Each block consists of three high temperature fission chambers. This system is expected to have a smaller response time and higher sensitivity due to its proximity in the core. Three failed fuel localization modules are also provided to identify the failed subassembly [6].

2.3. Experience with sodium systems [1–7]

Sodium systems have been operating for the past twenty-six years and their performance has been excellent. The impurity levels in sodium was always <0.6 ppm and it was demonstrated that even without a purification system in service for about 60 d, the impurity levels in the primary system remained within limits. During commissioning of the SG, one cold trap in the secondary sodium loop had to be replaced due to impurity loading at the time of connecting the SG to the loop. One secondary sodium pump was replaced after 10 000 h of operation due to abnormal noise. Performance of all other pumps up to now has been very good. There have been no incidents of oil leakage from the pump seals to the sodium circuit so far. Performance of the sodium pump drive system was not satisfactory initially. It improved significantly after air conditioning the control logic panels and carrying out certain logic modifications. The primary sodium was sampled periodically for trace element analysis and the nuclear grade purity was found to be well maintained. An electrochemical carbon meter has been installed in one of the secondary sodium loops on an experimental basis to measure the active carbon level in the system. The active carbon content was found to be within the limits.

There were incidents of ingress of mercury from the relief pot of the primary cover gas system into the primary sodium system during the vacuum pulling operation of the primary sodium storage tank and CRD compressor operation with a stuck closed NRV. This problem was overcome by modifying the layout of the relief line from the primary sodium storage tank and directly connecting the CRD compressor discharge line to the effluent header. In the PFBR, catch pots are provided to collect the mercury in case of carry over [6].

2.3.1. Primary sodium leak incident [1]

In April 2002, while the reactor was operating at 17.4 MW·t, there was a leakage incident involving 75 kg of primary sodium from the purification circuit. The leaked sodium froze on the cabin floor and pipelines and was manually cut and scooped out under inert purging. Leakage was from the body of a 20 mm size bellows-sealed valve, through one of the three blind holes used by the manufacturer for machining the valve body (Fig. 4). The valve was replaced. Since the problem is generic to the specific make, valves of this make used in the plant were inspected and defective similar valves were rectified by welding tight fitting plugs. The sodium which leaked during the incident was converted to hydroxide, neutralized by orthophosphoric acid and disposed off as active liquid effluent. It is gratifying to note that a material thickness of just 0.1 mm

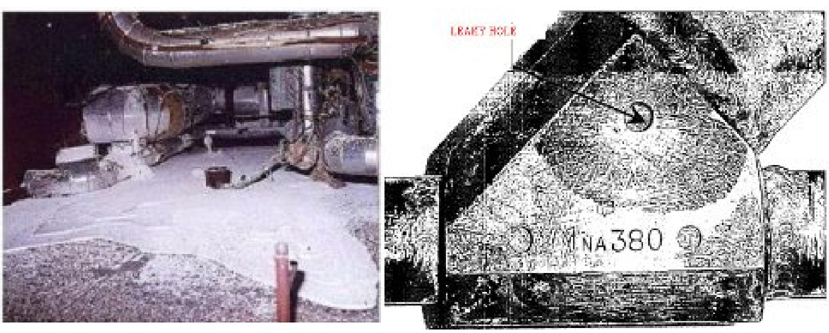

FIG. 4. Sodium leak inside purification cabin due to hole in the valve body.

was enough to hold sodium for 17 years. Experience with this method of disposal has been quite satisfactory and can be employed in the PFBR while disposing of sodium. Based on the experience of this incident, the PFBR has a more rigorous QA and inspection at the procurement stage for all sodium service valves.

2.3.2. Sodium leak from nickel diffuser into the vacuum circuit of the SG leak detection system of the west secondary sodium system [1]

In February 2006, there was a reduction in sodium flow in one of the SG leak detection circuits (Fig. 5). Heavy accumulation of frozen sodium was found in the shell side and vacuum line of the nickel diffuser. Sodium had leaked from the tube-to-tube sheath weld joint of the nickel diffuser and froze in the shell side and pipelines. This compressed and flattened the nickel diffuser tubes, resulting in low flow. As no sodium leak detection system was provided in the vacuum lines, the leak had gone unnoticed.

Hence, a mutual induction type probe was provided downstream of the nickel diffuser in the vacuum line to detect any leak in the initial stage itself. In the PFBR, electrochemical hydrogen meters are used for SG leak detection, in addition to the nickel diffuser and its associated vacuum circuits.

2.3.3. Modification in the surge tank level maintenance circuit

In the secondary sodium main circuit, the surge tank remained connected to the expansion tank through a communication line with a motorized valve and a bypass line across it (Fig. 6 (prior to modification)). A continuous flow of hot sodium from the surge tank to the expansion tank maintains this line hot and is available for communication whenever needed. As the hot sodium from the surge tank mixes with the cold sodium in the expansion tank, there was a possibility of thermal shock in the expansion tank where the hot sodium is mixed with cold sodium. An identical arrangement existed in French reactor Phenix and

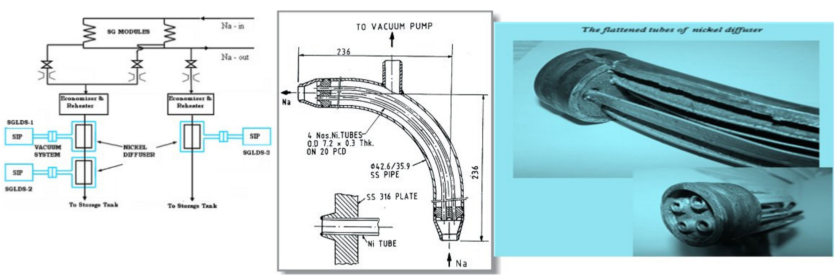

FIG. 5. Triplicated SG leak detection system and nickel diffuser showing flattened tubes.

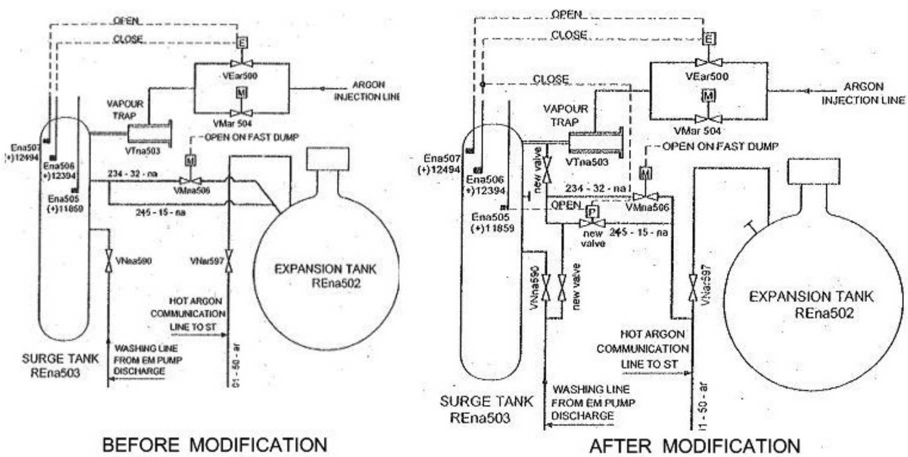

FIG. 6. Modifications in the surge tank.

leaks were reported at this location due to thermal striping. In order to prevent the recurrence of such a problem in the FBTR, the sodium communication line with the integral bypass line was converted to a hot argon communication line by modifying the circuit with a pneumatically operated valve, which will open whenever communication is required for level maintenance or for fast dumping (Fig. 6 (after modification)).

2.3.4. Replacement of rupture disc assemblies in the secondary sodium system [1]

Rupture disc assemblies are provided in the inlet and outlet sodium headers of SGs and in the cover gas region of the expansion tank to protect the SG and IHX bottom tube sheet during a sodium–water reaction. As per the technical specifications for FBTR operations, the rupture discs in the secondary sodium system are to be replaced periodically. As the FBTR was operating at a low sodium temperature and the purity level of sodium has been maintained extremely well, there was no concern regarding the corrosion effect at elevated temperatures and hence there was no need to replace the rupture discs.

However, the regulators recommended replacement of the rupture discs in one loop and subsequent burst testing of the removed rupture discs in order to ensure that there is no deviation in the set value due to ageing of the material. Accordingly, state of the art scored type rupture disc assemblies were procured and replacement was made in the secondary west loop. The removed rupture disc assemblies were burst tested and found to be rupturing at the design pressure, even after 25 years of service.

2.3.5. Choking of the hot argon line in the primary sodium system

Choking of the hot argon communication line of primary capacities was encountered a few times in the nozzle portion of the primary sodium storage tank due to deposition of sodium vapours in the cooler region. The cause of the loss of communication between the storage tank and the other primary sodium capacities is due to sodium deposition and choking near the storage tank nozzle (Fig. 7).

During reactor operation, the communication valve between the storage tank and the other primary capacities is in an open condition and communication is maintained between the storage tank and all other primary capacities. The sodium in the storage tank is in a frozen state and also the tank shell is at room temperature, except for the nozzle in the storage tank to which the hot argon line joins. The cover gas reject of the primary capacities is connected to the gaseous effluents through the storage tank. Sodium vapours travelling along this path are deposited in the colder region near the nozzle portion and block the argon communication between the primary capacities and the storage tank.

The choke was removed by cleaning after cutting the line. In order to keep the nozzle portion hot, heaters are laid in and around the nozzle portion. Also, the humps and bends in the hot argon communication line were removed.

2.4. Experience with the reactor assembly [1–7]

2.4.1. Deflection of reactor vessel during commissioning

During commissioning in 1985, when the sodium temperature was progressively raised to 350°C for isothermal tests, a large azimuthal temperature difference (~80°C) in the cover gas region of the reactor vessel was noticed. This resulted in tilting of the reactor vessel and a shift in the grid plate as measured by the displacement measuring device. This was investigated as being due to non-uniform natural convection currents in the cover gas space. This was overcome by injecting helium into the argon cover gas to form a double layer

FIG. 7. Argon line choked sodium.

above the sodium level to suppress rising convection currents. This works on the simple principle that heavy argon, on being heated by the hot sodium, tries to rise up and encounters the less dense helium at the top and cannot rise any further. The convection is only localized. This was found to be effective in bringing down the grid plate deflection and the temperature gradient in the reactor vessel.

In the PFBR, the annular gap between the main vessel and the roof slab is filled with wire mesh to avoid cellular convection of argon and associated circumferential temperature differences in the main vessel. Cellular convection in the annular gaps of the top shield due to penetration of various components such as IHX, PSP, CP, SRP, LRP, DHX, etc., has been studied both experimentally and theoretically and circumferential ΔTs were found to be within limits. Since the convection currents in the space above the sodium are expected to be weak, the helium injection provision is not required in the PFBR [6].

2.4.2. The fuel handling incident [3]

During an in-pile fuel transfer for performing a low power physics experiment in May 1987, a major fuel handling incident took place. The incident was due to a plug rotation logic remaining in a bypassed state during fuel handling, resulting in the rotatable plugs being rotated with the foot of a fuel subassembly protruding into the core during the transfer. Reactor operation could be resumed only in May 1989. A mechanical swivelling lock, to keep the subassembly firm in the transfer position, has since been installed in the transfer flask. This condition has also been wired to the logics of plug rotation and fuel handling flasks.

2.4.3. Water leak from BSC coils inside biological shield concrete [3]

The reactor vessel of the FBTR is surrounded by a safety steel vessel and further by two types of concrete, namely, a 600 mm thick biological shield and 900 mm thick structural concrete. A gap of 30 mm is provided between the two concretes to take care of differential thermal expansion. The biological shield concrete is cooled by circulating water through 180 coils embedded in the concrete.

In August 2000, an A3 subheader in header-A developed a leak. The leaking subheader was isolated. Water collected inside A1 cell (gap between steel vessel and biological concrete) was drained and reactor operation continued. There was no increase in the biological shield concrete temperature as coils from header-B cool the affected sector. In May 2001, the B5 subheader in header-B, which cools the same 60^{o} sector in the southwest portion of the concrete developed a leak. Identification of the leak location by water manometer tests and air purge methods indicated that the leaks were from socket welds in the coils. Crevice corrosion at the socket welds was considered to be the cause. Hence, proprietary formulation sealants were injected to arrest the leak points in the coils of the two subheaders and the coils were tested for their integrity. For the remaining coils, a global sealant treatment was carried out to arrest micro-leaks and the system was normalized and power operation resumed. Following these incidents, there were six more leakage incidents in different coils and these were chemically sealed. After arresting the leaks, the leakage rate from the BSC system has been reduced significantly, to within acceptable limits. On the basis of FBTR experience, the BSC coils in the PFBR have been designed and erected with butt welds.

2.4.4. Malfunctioning of core cover plate mechanism [1, 3]

About 95% of sodium flow through the subassembly exits through the openings at the top of the shoulder and a small flow of sodium jets through the 6 mm passage in the head and this is directed to the core thermocouples by sleeves on the core cover plate mechanism mobile plate. Since the jet is of small size, sleeves are to be positioned close to the head of the subassembly for guiding the individual stream for accurate temperature measurement.

During normalization of the pile after a fuel handling operation in July 1995, the core cover plate mechanism could not be lowered to its normal working position from the fuel handling position and became stuck 80 mm above the top of the subassembly heads. Though in the current position, the core cover plate mechanism does not pose any problem for normal operation or fuel handling directly, the temperature measurements of the sodium from the fuel subassembly in the outer rings (especially in rings 3 and 4) are not accurate (Fig. 8). The design provisions, such as radial entry of sodium flow into the subassembly and the high purity of sodium maintained, rule out blockage of flow through the subassembly. Sufficient margin is available to scram the reactor safely in the case of a flow blockage in fuel subassembly.

In the PFBR, core cover plate is a fixed part of control plug. Control plug holds the thermowells in which the core thermocouples are kept at around 100 mm above the top of the SA during normal reactor operation. The entire sodium flow through the SA exits through the top of the SA and since the diameter of the outlet stream is large (110 mm), the sodium streams will positively envelope the thermocouple located at 100 mm above SA head. Thermal hydraulic analysis indicated that the thermocouples of all fuel SAs are submerged in their respective streams and thus providing accurate temperature measurement [6].

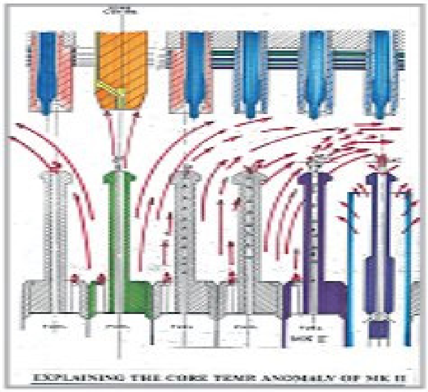

FIG. 8. Schematic of mixing of streams with core cover plate mechanism at 80 mm.

2.5. Experience with SG leak detection system [1]

The surface type reheater that preheated the sodium to 450°C in a nickel diffuser failed due to high sheath temperature of the heater during the commissioning stage. Hence, it was replaced by an immersion type reheater after which the performance of the reheater was satisfactory. During initial commissioning, it was noted that the response of the system was good at sodium temperature of 250°C and above and it was poor at sodium temperatures of less than 250°C. To take care of protection of SGs in the temperature range 180–250°C, a hydrogen detector in argon (HAD) was installed in the cover gas space of the expansion tank and commissioned. The response of the detector was found to be good in the temperature range of 180–300°C, during calibration of the system. After putting the detector into regular operation, the temperature of sodium for admission of feedwater to the SG was restored back to 180°C.

The life of filament of the mass spectrometer of the SG leak detection system was about a year. On many occasions, the reactor had to be shut down due to failure of the filament of the mass spectrometer. Sputter ion pump current, which is a measure of total pressure in the vacuum system, responded in a manner similar to the mass spectrometer signal during calibration of the system. Hence, the sputter ion pump current, instead of the mass spectrometer signal, was used to initiate reactor trip in the case of feedwater leak in the SG. With this, the need for periodic replacement of the filament of the mass spectrometer and consequent intervention in the vacuum circuit and reactor downtime were eliminated. The SG leak detection system is a complex system having an ultra-high vacuum system and complicated electronic signal processing circuit. On several occasions, there were spurious ‘spikes’ in the signal leading to reactor trip. It was a laborious and time consuming process to prove that the signal was spurious. Hence, the system was triplicated and trip initiated on two out of three coincidence logic. With the triplication of the SG leak detection system and various improvements carried out as above, the system became more reliable and there were no further spurious trips from the system.

Excess hydrazine dosing in the feedwater for dissolved oxygen control increased the hydrogen concentration levels in the sodium resulting in a spurious alarm. It is desirable to monitor the hydrazine level in the feedwater of the PFBR continuously to facilitate investigation in the event that the alarm is actuated. In the PFBR, the electrochemical hydrogen meter will be used for monitoring individual SG outlet hydrogen concentration and the nickel diffuser vacuum system will be used for the common outlet.

2.6. Experience with steam–water system

2.6.1. Replacement of contact type heaters by surface type heaters

In the earlier condensate system, direct contact type low pressure heaters with the integral hot well were used for their simplicity, better de-aeration, low cost and better efficiency. However, use of this kind of heater ended up in a loss of plant availability factor (on average 3–4 reactor trips per year) due to hot well level fluctuations and subsequent tripping of the pumps. The direct contact type heaters in the condensate system were replaced with surface type heaters and the condensate booster pump was also eliminated. After this modification, there was no incident of reactor trip from the steam–water system. In the PFBR, surface type heaters are used.

2.6.2. Seizure of main boiler feed pump

In April 1992, while preheating the feedwater system, abnormal noise was heard to be coming from the pump, which seized up. Investigation revealed that the failure was due to cavitation. The net positive suction head (NPSH) available to the pump was found to be close to the required NPSH and it further reduced during operating transient. The following modifications were carried out to improve the NPSH available.

- The MBFP leak off-line was rerouted to steam space of de-aerator with orifice near to the de-aerator to prevent heating of water at MBFP suction.

- Continuous cold water injection at $3m^3/h$ at MBFP suction to improve the available NPSH.
- Incorporation of additional recirculation line capable of handling $30m^3$ to avoid pump operation at low flow.
- In addition, instruments such as the recirculation flowmeter, pressure gauge in leak off-line and thermocouple in leak off-line and thrust bearing were incorporated.
- Feedwater heating procedure was modified by using steam from package boiler instead of pumping power of MBFP.

With these modifications implemented, MBFP operation was satisfactory.

2.6.3. Orifice dislocation in SG

During the operation at 18.6 MW·t in the 15th irradiation campaign, a large variation was observed among the steam temperatures from the four SG modules. The bulk steam outlet was also only 430°C, as against 450°C estimated. Investigation revealed dislocation of the spring loaded orifice assemblies provided at the entry of the water tubes for providing flow stability to the SG modules (Fig. 9). It was seen that the orifice assemblies from 22 tubes out of the total of 28 tubes had dropped from their positions. All the orifice assemblies were replaced by a welded design and the SGs were normalized in December 2009. During subsequent operation, flows through all the modules were found to be equal and steam temperature variations were within limits.

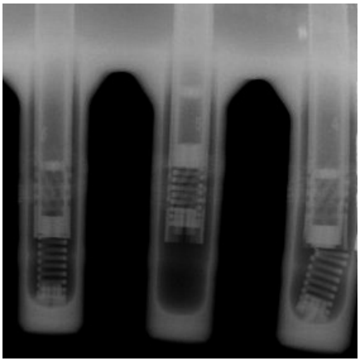

FIG. 9. Dislocated orifice assemblies in the tubes of SG modules.

2.6.4. Blanking of three tubes in each SG

The sodium outlet temperature up to 2007 was only about 420ºC, except for a brief spell at 444ºC in 2002. It was decided in 2007 to blank three out of seven water tubes in each SG module in order to achieve operating temperatures close to the design temperatures at the power levels realizable with the current core size (~22 MW·t). This modification was carried out in 2008. A maximum sodium temperature of 490ºC was realized while the turbogenerator was in operation generating 4.2 MW(e).

3. CONCLUSIONS

The FBTR has been in operation for the past 26 years. The experience and confidence gained with the FBTR has enabled us to leapfrog to a stage where currently about 1400 t of sodium has been charged into the storage tanks of the PFBR without any incident. Starting from the NaK leak incident during commissioning, every significant event in the FBTR has had its prompt feedback into the PFBR design. Every incident is studied by the designers for its applicability to the PFBR. It is analysed for its implications on PFBR operation. Wherever required, the design is modified. The successful operation of the FBTR, the excellent track record of nuclear and radiological safety and the confidence in the design based on FBTR feedback experience are some of the main factors in according the project sanction for the PFBR.

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EXPERIENCE AND SOLUTIONS IN THE DECOMMISSIONING OF SODIUM COOLED FAST REACTORS

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Abstract

In a period of active employment of the new kind of energy — nuclear power — at various industrial plants, in technological processes, transportation units, in medicine, for R&D purposes, in the defence sphere, the problem of decommissioning the nuclear and radiation hazardous facilities (NRHF) has not been given adequate attention. To date, the problem has become a pressing challenge. According to predictions, by 2030, the decommissioning of 30 A-units is planned in the Russian Federation, as well as 13 defence reactors, more than 30 research reactors, several large enterprises of nuclear fuel cycle and open storage ponds for liquid radioactive wastes. A similar problem has to be resolved for the nuclear prototypes — the research and power reactors with fast neutron spectrum. The process of decommissioning the NRHF, in addition to financial problems to be resolved, necessitates a certain improvement in solutions in the administrative and legislative aspects, as well as on problems in science and technology. The latter are caused by sparse experience with the decommissioning of NRHF, great diversity of their engineering design solutions and physical characteristics, the lack of technologies needed, robotics, infrastructure for the dismantling of equipment with high radioactivity levels, conditioning of the radioactive wastes, reprocessing of spent nuclear fuel, and storage and final disposal of radioactive wastes. The two current approaches considered in the world for the management of the entire process of NRHF decommissioning, namely, that of delayed dismantling, and an option with immediate dismantling. For the time being, the Russian Federation has used the first of the approaches mentioned. The domestic and foreign experience, not extensive as it is, gained from the decommissioning of fast reactors with sodium (sodium-potassium) coolant has been highlighted in the report.

1. INTRODUCTION

Academician V.I. Vernadsky wrote, in 1922, before discovery of the neutron and nuclear fission “We are approaching a great breakthrough in mankind’s life, not comparable with anything known before. The time is not very far ahead, when man can manage nuclear energy himself, a source of force to give him a possibility to organize his life at his own will. This can happen after some centuries; however, it is clear that this must come into being. Will man be able to use this force, direct it for the purposes of goodwill, rather than to self-destruction? Has he become mature enough to be capable of using the force to be inevitably provided by the science?” [1].

His prediction became true very soon, his apprehensions becoming real as well, to a certain extent. From the very beginning, the use of nuclear energy was aimed at obtaining nuclear weapons. It was only after its employment by the United States of America in the war with Japan and development of similar weapons in the former USSR, that humankind started applying nuclear energy, in a intense and steady way, in areas such as power generating reactor facilities to produce thermal and electrical energy, desalination of seawater, research reactors, power installations of submarines and surface vessels, and spacecraft.

During that period of development and creation of nuclear weapons and the initial period of R&D and construction of power and research reactors, no proper attention was given to the decommissioning of the nuclear and radiation hazardous facilities (NRHFs), either in the Russian Federation, or abroad. However, to date, the problem has become a pressing challenge, far from safe, and extremely expensive. As early as 2006, operation of 109 NRHFs was discontinued, including 65 A-submarines awaiting utilization. According to the prediction plans, by 2030, about 30 nuclear power units will be subject to decommissioning, as well as 13 commercial reactors for weapons grade Pu production, more than 30 research reactors, a number of large nuclear fuel cycle enterprises and open storage pools for liquid radioactive wastes [2].

A similar problem will have to be resolved also for research and power reactors with a fast neutron spectrum, being pioneers in the strategic area of nuclear power, including 8 research reactors shutdown in France, Germany, the Russian Federation, the United Kingdom and the United States of America, and 6 power reactors in the France, Germany, Kazakhstan, UK and USA. In the forthcoming years these may be joined by research reactor BOR-60 in the Russian Federation, JOYO in Japan, as well as power reactors MONJU, also in Japan, and BN-600 in the Russian Federation.

2. OBJECTIVES, PROBLEMS AND STRATEGIES

Decommissioning of an NRHF naturally suggests that its operation will be discontinued; however, termination of operating a facility does not mean its decommissioning in a certain aspect. The IAEA documentation defines a series of managerial and engineering procedures as the decommissioning, aimed at the withdrawal of a site (facility) from the supervisory bodies’ domain with a possibility to use the land plots and buildings for other purposes. The Rostechnadzor documents state as follows: “Decommissioning of a power unit at the NPP includes the activities that follow the removal of nuclear fuel and nuclear materials from the NPP unit, aimed at the attainment of a pre-specified end state and condition of the NPP unit and precluding the employment of the said unit as a source of energy" (NP-012-99) and "Decommissioning of a research nuclear facility (RNF) includes the activities after removal of nuclear materials from the RNF site, aimed at the attainment of a pre-specified end state and condition of the RNF and its site" (NP-028-01). The definition of decommissioning as a notion stated in our regulatory documentation is considered to be preferable for our conditions and makes it possible to implement an optimal variant of decommissioning in each specific case.

The regulatory documentation of Rostekhnadzor stipulates three possible ways for decommissioning of an NRHF: (i) liquidation of NPP (RNF) with immediate dismantling of all radioactive equipment, disposition of all radioactive wastes from the site and writing-off from the regulatory authorities' supervision (the greenfield site state); (ii) preservation of the NPP (RNF) status under supervision, with delayed dismantling of radioactive equipment in order to obtain partial decay of radioactive elements, the subsequent dismantling thus made easier; (iii) burial of NPP (RNF), the dismantling being delayed for a long time (or passing over the dismantling step), taking measures for the environment protection against hazardous radiation effects caused by the NPP (RNF) being decommissioned and organizing appropriate monitoring. Taking into account that our infrastructure is not prepared, our legislative and regulatory fundamentals being immature, and funding insufficient, for the time being the option of preservation under monitoring is accepted as the main variant for NRHF decommissioning. The European countries, proceeding from the analysis of experience gained, arrive at the conclusion that the first option is preferable for the NPP (RNF) decommissioning, as a cheaper and safer one. According to the recommendation documentation of Rosatom, the decommissioning should be fulfilled in three steps: (i) preparatory step (with a preliminary sub-step), (ii) step for preservation under monitoring (with sub-step of preparing for the preservation under monitoring) and (iii) the activities directly related to the dismantling of highly radioactive equipment, buildings and structures, moving out the solid and liquid wastes preliminarily conditioned, wherever necessary, and possible utilization of a part of the wastes, equipment and structures after cleaning them and implementing decontamination and control (verification) procedures.

The works for decommissioning of an NPP or RNF that used fast neutron reactors will repeat, in many respects, the organization, schematic planning, procedures for decommissioning of facilities with thermal neutron reactors. However, they will considerably differ in terms of solutions for the technological and engineering problems, owing to the presence of sodium (sodium-potassium) coolant.

During the preliminary part of the preparatory step to be realized several (five) years prior to the reactor shutdown, a principal programme for the decommissioning is developed, including the concept, the initial design documentation is prepared with all modifications and the history of facility operation with information on all transients/accidents, after which the order on facility decommissioning is issued. The first preparatory step should also include the following activities:

- Development and implementation of the programme for complex engineering and radiation survey of the facility;
- Reactor shutdown and bringing the facility to the nuclear and radiation safe condition, including unloading of nuclear fuel from the reactor, cleaning it of sodium, and removal of the fuel beyond the site;
- Removal of radioactive sources, operation media, collected wastes from operation, reprocessing of radioactive liquid metal coolants (sodium and sodium-potassium), with their conversion into a fireproof/flameproof state suitable for long term storage;
- Development of technological processes (flow processes) for conditioning of liquid metal coolants, cold traps for oxides, caesium traps, cleaning the circuits, equipment, including the reactor, pumps, intermediate heat exchangers, steam generators, storage tanks for coolant, etc.;
- Development of the project for decommissioning of the facility and obtaining approvals needed.

In accordance with regulatory documents, activities in the frame of the first step are implemented on the basis of the licence valid for the normal operation conditions, regulations and guidelines for the so-called ‘final shutdown’ regime. In reality, many of the first step tasks mentioned above may prove to be a novelty and difficult to conduct, even for experienced personnel.

The licence for decommissioning can be issued only after the first step tasks have been performed; the fuel transported away from the facility; after removal of the operation media, sources; collection of the RW; and after approval of the project for its decommissioning. The following works are performed during the preparatory part of step 2:

- Repeated analysis of the building structures’ condition, that of systems and equipment to be used during the step of preservation under monitoring and, if necessary, the cables, instruments, ventilation system equipment are replaced, as well as that of lighting, heating, water supply, cleaning and decontamination, etc., where the service life expired.

- In accordance with the decommissioning project, the toolboxes are fabricated, test facilities are constructed, as well as storage facilities and warehouses necessary for the RW conditioning, dismantling and temporary storage of the equipment subject to dismantling.
- The dismantling of non-radioactive and low radioactivity level equipment is fulfilled.
- The equipment, circuits, reactor coolant drain tanks and reactor are cleaned to remove the residues of sodium not drained, and are then dismantled (except for the reactor).
- The tasks for conditioning of cold traps for oxides, traps for sodium vapours and caesium traps are fulfilled.
- The state of the protection barriers and tools for control of active substance dissemination beyond the barriers is monitored and, where necessary, additional protection barriers are installed, as well as the structural means for control of radioactive substance release.
- The tasks are fulfilled for localizing the equipment with high radioactivity levels which remain in place for storage.
- The new in-line documentation (or forms and records) is developed for the period of storage (cooling) under monitoring.

The period of 50–70 years is usually considered an acceptable cooling period. At this step, it is vital to ensure the physical protection, safety and performance of the structures and equipment, monitoring of the protection barriers and release of radioactive substances, as well as the preservation of design documentation.

At the final step, all remaining equipment is dismantled, including the reactor facility, with dismantling of the biological protection devices, buildings and structures, all wastes are moved out to the temporary storage or final disposal sites. In order to fulfil this work, an additional project is implemented for the facility decommissioning, and special equipment is designed and manufactured preliminarily (robotics, etc.)

The experience really gained with decommissioning of NPPs and RNFs with thermal neutron reactors, and particularly with fast neutron reactors, is apparently too small, and therefore one cannot consider the steps described above and the sequence of work performance as being well established. The intensifying of NRHF decommissioning is still hindered by three problems awaiting their ultimate resolution:

(i) The problem of funding;
(ii) The problem of administrative, legislative and regulatory issues;

(iii) The problem of engineering development of optimal technologies, special instrumentation, and technical devices for the dismantling works and for transportation of the high level wastes, and the State storage facilities and sites for burial of the wastes and production capacities for the reprocessing of irradiated nuclear fuel.

3. STATUS OF PROBLEM SOLUTION AND EXPERIENCE GAINED

The decommissioning of NPPs and RNFs has proven to be a time consuming and costly matter. The IAEA has determined that in the mode of decommissioning of NPP serial units, as formed now, the process should last 15 years and cost should be about US $350 million (2003 price levels) and for the research nuclear facilities $ US1 million/MW and 3 years, respectively, plus the preparatory period.

According to the estimations of Commissariat a l'Energie Atomique et aux Energies Alternatives (CEA), the cost of decommissioning NPP units is €200/kW(e). The prediction estimations of our experts show that decommissioning of NPP units amount to ~15–18% of the total capital expenditure, which coincides with estimations made by the IAEA and the CEA.

TABLE 1. ESTIMATED DECOMISSIONING COSTS

Reactor/facility	Startup year	Shutdown year	Duration of decommissioning	Cost of decommissioning
RAPSODIE	1967	1983	2020	~€50 million
DFR	1959	1977	2026–2042	£250 million
EBR-II	1962	1994	?	?
BR-10	1959	2002	~2060 (2025)	~300 million RUR (2002–2011)
FFTF	1980	2003	?	?
Fermi	1963	1973	?	?
PFR	1974	1994	?	?
BN-350	1973	1999	?	?

TABLE 1. ESTIMATED DECOMISSIONING COSTS (cont.)

Reactor/facility	Startup year	Shutdown year	Duration of decommissioning	Cost of decommissioning
Phenix	1973	2009	2029	€900 million
Superphenix	1986	1998	2035	€1.5 billion

In reality, currently both costs and duration of decommissioning exceed the figures cited; this is especially the case with the new experimental power units. Thus, for example, even though the first power unit of the Beloyarskaya NPP was finally shut down in 1981, nevertheless, the tasks of the 1st step have not been finalized yet and the licence for decommissioning has not been granted. In the process of decommissioning of an experimental high temperature gas cooled reactor with ball-shaped fuel elements, AVR (15), the costs of work for its decommissioning have increased more than 20 times, from €20 million to €490 million, which probably exceeds the total capital input for the creation of this facility.

The situation with decommissioning of experimental power reactors and fast neutron research reactors can be illustrated using Table 1. As Table 1 shows, the duration of the decommissioning of fast power and research reactors ranges within 20–60 years, and its cost nowadays, in some cases, exceeds the total capital input for the facility's creation. Taking into account the enormous expenditures, it was necessary to find a definite source of funding. According to governmental decree No. 367 of 02.04.1997, the operating entity must establish a special fund as a source for financing the decommissioning NPP units, using the special allowance included in the cost of production (electrical and thermal energies), whereas the decommissioning of research facilities must be funded from the budget. The Federal target programme, Assurance of Nuclear and Radiation Safety for 2008 and for the Period to 2015, approved by the governmental decree No. 444 of 13.07.2007 stipulates funding for creating the related infrastructure (storage facilities for RW, production facilities for storage and reprocessing of spent nuclear fuel, etc.) and for decommissioning of NRHF.

The insufficiency of the administrative, legislative and regulatory basis is one more problem for the decommissioning of NRHF, especially in the aspect of handling/management of RW. A legislative based solution is needed, such as categorization of the RW, requirements governing storage facilities' holding of low, intermediate and high level wastes, responsibility for storage of RW, funding of the construction and operation of storage facilities, feasibility of final disposal of liquid RW in geological strata, the entities proposed as owners of the RW disposal sites, aspects of management of diverse kinds of spent nuclear fuel, etc.

The Federal law signed by the President of the Russian Federation on the management of radioactive wastes has, to a certain extent, resolved the problems listed. In particular, the law determines that:

— The sites, facilities for disposal of RW will be in Federal ownership, or otherwise they can be owned by the SC Rosatom.
— The radioactive wastes containing nuclear materials will be exclusively in Federal ownership.
— The national operator (the legal person carrying out activities for the management and burial of RW) will be determined by the Government.
— Safety liabilities in the management and disposition of RW, until they are transferred to the national operator, will lie with the organizations where the wastes are produced.

The law determines the sources for funding the tasks for management of RW and defines the requirements for the storage, final disposal, state accountancy and supervision of the RW storage. In particular, it has been determined that no construction of new sites for disposal of liquid RW in geological strata will be allowed. Besides, it is prohibited to import any RW on to Russian Federation land for the purposes of storage, reprocessing and final disposal. The main trends for preparing a special regulatory basis and creating a system for final disposal of low and high level wastes have been defined.

Another problem for the decommissioning of NPPs and RNFs with BN-type reactors is associated with the need to develop new technological processes for the conditioning of radioactive coolants and of the related equipment (cold traps for oxides, caesium traps, pumps, heat exchanging equipment, reactor vessel and reactor internals), pipelines and tanks; and using these processes as the basis for creating facilities for realizing these processes for conditioning the components listed above. Besides, it is necessary to wash-out spent fuel assemblies from residues of coolant, including those with defective fuel elements, if that had not been fulfilled in the course of facility operation. The UO_2 fuel of the BN-350 and BN-600 facilities was washed out using the slightly superheated vapour mixed with nitrogen; in the form thus obtained, washed from sodium residues, it was kept in the storage ponds until shipment to the RT plant, where it was subsequently reprocessed, with the preparation of regenerated U and Pu. Fuel from BR-5 (BR-10), UO_2, PuO_2, UC, UN, was basically stored in the form and not washed off from sodium, in tight cases. Therefore, actually, its cleaning by a conventional vapour–gas method had already been started in the process of facility decommissioning. It is logical to be concerned that the vapour–gas technology might appear infeasible for the cleaning of spent fuel assemblies with UC fuel containing non-tight fuel elements because of pyrophosphorous characteristics of the fuel. Besides, it is time to develop a technology for reprocessing the UC-fuel; this was suggested to be a development of NIIAR.

The process of cleaning the primary circuit of the BR-10 reactor facility was fulfilled three times in the course of facility operation and once during the decommissioning. In the latter case, not only was the primary circuit subjected to washing off, but the auxiliary systems as well, except for the cold trap for oxides, including the spectrometry section, sampling device for sodium and central loop channel as well. The sequence of cleaning procedures was as follows:

(i) Drainage of sodium from the circuit;
(ii) Vacuum removal of sodium residues in the circuit at temperatures of 380–450°C and a vacuum of 0.1–1.0 mmHg;
(iii) Vapour–gas treatment;
(iv) Treatment with $KMnO_4$ solution;
(v) Washing with distillate;
(vi) Treatment with oxalic acid solution adding 1% solution of hydrogen peroxide;
(vii) Washing the circuit with distillate and drying it.

In the vapour–gas treatment, monitoring of hydrogen was provided; the vapour supply was stopped when concentration of hydrogen gas reached 3%. A decrease of the γ dose rate in the primary circuit premises of over 100-fold was achieved by this cleaning.

Similar technology was used in the Russian Federation for cleaning large size equipment from residues of sodium (i.e. vapour–gas treatment followed by washing with water), e.g. steam generators at the BN-350, and pumps and intermediate heat exchangers at the BN-600. The design firm OKBM used the same technology when cleaning the circuits and equipment of experimental facilities from sodium, and the IPPE applied it to the cleaning of cold traps for oxides. An improved variant of cleaning the equipment with ‘heavy spirits’ (butylcellosolv) was tested for these applications in France, the USA, and the former USSR. However, explosions of this mixture occurred when these technologies were tried in the USSR (IPPE) for cleaning one of the traps for oxides using the vapour–gas and water method, as well as in France (Cadarache), when ethylcarbitol was applied for cleaning the reactor coolant drain tank at RNF Rapsodie.

In order to develop a preferable variant of the cleaning method for tanks, cold traps and other equipment of BN reactor facilities, a new method of cleaning was developed at the IPPE using nitrous oxide (N_2O), which was tested after laboratory experiments fulfiled, first, in the course of cleaning of one of the loops of BR-10 secondary circuit, and then on the cold trap for oxides in this circuit. The operation for cleaning proceeded as follows:

(i) Heating followed by drainage (pressing out) of sodium from the circuit (cold trap for oxides);
(ii) Oxidation of sodium remaining in the circuit (cold trap for oxides) by cyclical pumping of a mixture of argon with nitrous oxide through it;
(iii) Dissolution of the product obtained in water and drainage of the solution.

The process was monitored by measurement of N_2O, N_2, O_2, H_2 contents in the samples taken on a gas chromatograph. The analysis of a macroporous brittle product formed as a result of oxidation showed that on average it consists of 21% Na_2O, 44% $NaNO_2$, 22% $NaNO_3$, and 12% $Na_2N_2O_2$. No hydrogen presence in the spent gas medium was detectable. Some work on making this product monolithic (single structure) was fulfilled.

French experts have developed and used another technology for cleaning the circuits and large equipment from residues of sodium: wet carbon dioxide. First, work to achieve maximum removal of sodium is done. This is done in order to reduce the time of operation that follows, treatment with gas, so that to ensure better accessibility of technical tools for dismantling the object equipment, for improving the process monitoring and enhancing safety.

The R&D done provided the information that under optimal conditions, at a temperature of 40°C, a layer of sodium 50 mm thick reacts completely with CO_2 (carbonized) for 1500 h; the volume of products formed being five times more than the initial volume of sodium. It was found out that in some cases, in closed or encumbered places, the carbonates are subjected to compression, their porosity decreasing, which can terminate the process of carbonization and lead to preservation of a certain part of non-reacted sodium. Upon completing the carbonization, the carbonates formed are removed by dissolving them with water, first by rinsing, then filling with water, with the hydrogen being under very careful control.

French specialists suggest this technology for the first step in the cleaning of large size equipment, including the reactor vessel. Then, it is proposed to cut the large size equipment into fragments, and finally wash with water. The reactor vessel after carbonization will be filled with water, dismantling of the reactor internals to be organized as a subaquatic procedure. After that, water will be removed, and the reactor vessel will be dismantled in the air environment using special remotely controlled equipment.

The liquidation, or to state it more precisely, conditioning of radioactive sodium will be fulfilled in France, Germany, the UK and the USA by a technology initially developed in France and the USA (the NOAH process). According to this technology, liquid sodium is injected under high pressure and at a temperature of ~180°C via the jet into alkaline water (40–50%), where the temperature must be maintained below the solution's boiling point ~138°C (for a 50% solution) by means of cooling, and maintaining the concentration of alkaline solution range within 40–50% by supplying water to the chemical reactor. Then, using carbon dioxide, the alkaline solution is transformed into soda, $2NaOH + CO_2 \rightarrow Na_2CO_3 + H_2O$, and one of the variants, radioactive soda, is transformed into a cementstone. The facility designed and assembled in France provides the processing of 5 t/d of sodium, which will allow the reprocessing of 5500 t of sodium at Superphenix over 1100 d.

To date, in the Russian Federation, the technology has undergone laboratory tests only. For the conditioning of sodium, specialists of the BR-10 have developed a technology for its oxidation and solidification (making monolithic) using slugs from copper smelting production. According to this technology, melt sodium is transferred to the chemical reactor filled with a slug, the mixture temperature being raised to 1100°C over 15–20 s, which ensures completeness of the process. After cooling, the mass reacted becomes a solid, stone-like material.

Conditioning of cold traps for oxides is one of the most difficult tasks owing to their high radioactivity levels, specific features of design and the spectra of the chemical and high radioactivity level substances accumulated, including caesium, Na-22 and tritium. With the objective of conditioning the cold traps for oxides, different methods were studied in the Russian Federation, such as the vapour–gas–water, water–alkali, vacuum, and gas methods (on the basis of nitrous oxide). The first and the last methods listed were used for washing the cold traps for oxides at the IPPE, the water method was applied for cleaning two traps at the NIIAR. French specialists propose cutting the cold traps for oxides into small fragments after removal of sodium and destruction of sodium oxides, hydrides and tritium, then washing them with water.

4. CONCLUSIONS

The conclusions can be summarized thus:

(a) Decommissioning of NRHF, including the NPP and RNF, has become a pressing challenge, expensive to resolve, and not definitely a safe problem.

(b) Three concepts have been considered in the practical works of different countries for decommissioning of NRHF: (i) with immediate dismantling ‘in total’, to the greenfield site state, followed by withdrawal from the domain of supervisory authorities, (ii) with a dismantling delay of 20–100 years and (iii) with on-site final disposal. Nowadays, decommissioning with delayed dismantling is considered as a priority for the Russian Federation. However, increasingly solid arguments have been postulated for immediate dismantling; it is cheaper and more justified from a moral perspective.

(c) The administrative and legal fundamentals for the solutions, procedures and technologies have been developed as ‘a first approximation’ only. It is necessary to gain and analyse the experience and define the optimum standard solutions and it is necessary for the country to resolve the problem of storage facilities for RW, provide the development of special remotely controlled robotics, create the sites and facilities for storage and reprocessing of spent nuclear fuel, and ensure financial support to solve the problems related to NRHF decommissioning.

(d) It is urgent to designate the persons and entities responsible for carrying out all activities for decommissioning NRHF.

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SKILL CAPABILITIES, PROFESSIONAL DEVELOPMENT, KNOWLEDGE MANAGEMENT

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EDUCATION AND TRAINING IN SUPPORT OF SODIUM COOLED FAST REACTORS AROUND THE WORLD

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Abstract

The Generation IV Technology Roadmap has identified six systems for their potential to meet the new technology goals to improve safety, sustainability, economic competitiveness and proliferation resistance. Among these systems, three are fast neutron reactors: two cooled by liquid metal, the sodium cooled fast reactor (SFR) and the lead cooled fast reactor, and one cooled by gas, the gas cooled fast reactor. The SFR has the most comprehensive technological basis as a result of the experience gained from worldwide operation of several experimental, prototype and commercial size reactors from the 1940s. In order to support the operation of existing reactors, design activities for new projects and decommissioning of old reactors, it is mandatory to maintain and develop skills, particularly among the young generation. This paper presents the current strategies developed at the national level, or within a multilateral framework such as the EU or the IAEA, to support the development of SFRs, with particular focus on education and training initiatives dedicated to students, researchers, designers and operators involved in the development of SFRs.

1. INTRODUCTION

The Generation IV Technology Roadmap has identified six systems for their potential to meet the new technology goals to improve safety, sustainability, economic competitiveness and proliferation resistance. Among these systems, three are fast neutron reactors: two cooled by liquid metal, the sodium cooled fast reactor (SFR) and the lead cooled fast reactor, and one cooled by gas, the gas cooled fast reactor.

In Europe, the Strategic Research Agenda of the Sustainable Nuclear Energy Technology Platform (SNETP grouping over 100 stakeholders from industry and research organizations) has selected these three fast neutron reactor systems to support the deployment of sustainable nuclear fission energy. Fast reactors' development needs an important technology support to finalize their innovative design and to assess their safety. The SFR concept is currently considered as the reference technology within the European Strategy framework.

Among fast neutron reactor systems, the SFR has the most comprehensive technological basis as a result of the experience gained from worldwide operation of several experimental, prototype, and commercial size reactors from the 1940s. This experience amounted to about 402 reactor-years of operation by the end of 2010. Six reactors are still in operation: the CEFR in China, the FBTR in India, Joyo and Monju in Japan, and the BOR-60 and BN-600 in the Russian Federation. Two reactors are being built: the PFBR (500 MW(e)) in India and the BN-800 (800 MW(e)) in the Russian Federation. Several projects are currently being developed: the CDFR in China, ASTRID in France, the CFBR in India, the JSFR in Japan, the PGSFR in the Republic of Korea and the BN-1200 in the Russian Federation. In order to support the operation of existing reactors, design activities for new projects and decommissioning of old reactors, it is mandatory to develop skills, more particularly among the young generation, who will operate these new reactors. In addition, education and training is essential to share the knowledge among teams involved in research and development. Several strategies are developed at the national level, or within a multilateral framework, such as the EU or the IAEA, to support the development of fast reactors. These education and training initiatives are generally focused on the following main tracks:

— SFRs and Generation IV requirements;
— Sodium properties and consequences on technologies;
— Core design, neutronics and physics;
— Fuel development;
— Material issues;
— Safety approach for SFRs;
— Modelling and simulation;
— Support for operation of SFRs;
— Decommissioning and dismantling.

Several countries have developed specific ‘sodium schools’, mainly in France, India and Japan, to support a major requirement for education and training focused on sodium technology.

These education and training activities are dedicated to students, researchers, designers and operators involved in the development of SFRs and related experimental facilities.

2. SODIUM PROPERTIES, CONSEQUENCES OF R&D TOPICS AND PRIORITIES IN EDUCATION AND TRAINING

Sodium is the most common of the alkali metals. It is widespread in nature but only in the form of compounds (sea salt, rock salt, carbonates, nitrates) and it is the sixth most abundant element in the earth’s crust (about 2.8% of terrestrial rock) and is present in great quantities in seawater as sodium chloride. Owing to its extensive use in industry, mainly for chemicals, electronics, etc., its cost is low.

In summary, sodium has the following main properties:

- Low melting point of 97.8°C, allowing shutdown and handling operations at temperatures below 200°C, i.e. 180°C, and avoiding risk of freezing in the steam generator units, particularly if the steam generator unit is used as a decay heat removal system. Moreover, this low temperature favours periodical inspection campaigns.
- Liquid over a large range of temperature (97.8–881.5°C). Usually, cores of SFR have a positive void coefficient which means that in case of absence of Na (i.e. boiling), reactivity is inserted, which can induce a power transient. For ASTRID, the CEA and its partners have designed a new core with an overall negative void coefficient if a boiling phase is reached.

- Low density and low viscosity. Owing to the similitude between sodium and water densities and viscosities, it is possible to carry out some experimental thermohydraulic studies and code validations with water. The low density of sodium favours passive and mastered fuel relocation by gravity in a core catcher, in case of core destruction, avoiding possible re-criticality. Low density also allows having passive shutdown systems, by gravity. Low density favours ultrasonic transmission in structures, owing to the large difference in density between steels and Na.
- Very high thermal conductivity of sodium and very attractive heat capacity properties.
- A coolant which does not slow down neutrons.
- Very limited activation, with short decay periods (^{22}Na: 2.6 years, ^{24}Na: 15 hours) and no alpha emitters (such as ^{210}Po).
- Very good compatibility with steels: no liquid metal embrittlement, very low corrosion kinetics, limited mass transfer and consequently low dosimetry, demonstrated by years of operation.
- Excellent electrical conductivity, allowing the use of specific technologies such as electromagnetic pumps and electromagnetic flowmeters.
- Very limited amounts of particles in sodium, owing to the instability of ternary oxides (except $NaCrO_2$) and high dissolution rates in Na, owing to its reducing properties.
- Owing to its low saturation vapour pressure, a very limited Na transfer in the cover gas plenum would occur, inducing few deposits in the upper structures. Moreover, in the case of fire, the sodium flames are very short and the heat produced by the fire is rather low. This allows a Na fire to be extinguished by spreading powder.
- Low oxygen and hydrogen solubilities in Na, almost zero near the melting point, allowing its purification thanks to the cooling and retention system called ‘cold trap’.
- Very good wetting properties, owing to the reduction of metallic oxides with Na over about 300°C. This property improves the quality of periodical inspections, carried out with ultrasonic systems and needed because of the opacity of the liquid metal.
- A very important reaction with water, which can induce deleterious effects in steam generator units in the case of a pipe rupture, but which allows component cleaning, prior to repair and Na treatment during decommissioning phase (conversion into sodium hydroxide then sodium salt, without any inherent toxicity). The sodium–water interaction in steam generator units has to be avoided or mitigated by design. It can be easily detected due to the production of hydrogen, which can diffuse through a nickel membrane and can be measured by a mass spectrometer, without

rupture of the confinement. This interaction can also be mitigated at the design stage, by selecting, for instance, a modular system. For cleaning pits or Na treatment processes, the risk of explosion due to hydrogen has to be mitigated by dedicated means such as inertization, recombiners and appropriate design of buildings.

- An important chemical reactivity with air, which can induce a Na fire. This event can be avoided by inertization or mitigated by early detection or confinement or by adding dedicated powder extinguishers.

To summarize, sodium is a good compromise, having very good thermal properties, low viscosity, low density, low activation by neutrons, good compatibility with materials, ready availability, being cheap, but its reactivity with air or water has to be mastered.

From these properties, it is easy to deduce some of the key issues to be described and discussed during education and training initiatives.

3. EDUCATION AND TRAINING INITIATIVES

3.1. China

The CEFR achieved criticality for the first time on 21 July 2010. The reactor went on to increase its thermal power from 9 February 2011. It began, for the first time, generating electricity to the grid at 40% of its nominal power for over 24 hours on 21 July 2011, and reached the acceptance objective of its leadership. Since then the reactor is, up to now, in a state of cold shutdown for conducting the inspection acceptance procedure used by the National Science and Technology Ministry and the Science and Industry Bureau of the National Industry and Information Ministry. The sodium in both primary and secondary are both kept at around 230°C and their plugging temperatures are maintained at about 105°C. The CEFR planned to increase the thermal power in December 2012 and to reach its thermal capacity of 65 MW in June 2013. Meanwhile, the 600 MW(e) Demonstration Fast Reactor is under design. Furthermore, the CEFR is one of the first 16 national energy (resource) research and development (and experiment) centres recognized by the National Energy Administration on 1 January 2010 and named Fast Reactor Research Centre.

In the next 5 years, new laboratory buildings will be set up, including a laboratory building for key equipment (F-202), another one for chemistry analysis and material research (F-203) and finally, one to study sodium fire, sodium–water reaction and waste sodium disposal (F-204). The objectives of research and demonstration of each building are as follows:

(1) F-202: A large scale sodium loop experimental platform will be set up in the building for acquiring the hydraulic characteristics of the sodium pump and for testing sodium pumps, sodium valves, sodium meters, etc.
(2) F-203: A building to analyse all kind of impurities in sodium, cover gas and water from the third loop, to study and test materials for fast reactors and performance of the latest materials in operation.
(3) F-204: A building to study sodium fire, sodium–water reaction, and treatment of sodium waste and its disposal. The causes, development mechanisms and consequences of a pool sodium fire, spray sodium fire and mixed sodium fire will be mainly studied. Programmes to fight against the sodium fires will also be developed. Research on instant monitoring, fireproof methods and release steps for the consequences of a pool sodium fire, spray sodium fire and mixed sodium fire will also be undertaken. Research on real time monitoring, fireproof methods and release steps for the consequences of a pool sodium fire, spray sodium fire and mixed sodium fire will also be performed. Monitoring and control methods for the reaction between sodium and water will be studied with a sodium–water reaction test facility. Radioactive sodium disposal processing and key technologies will be developed to study the reaction mechanism of sodium disposal processing, removal of reaction heat, absorption and removal of hydrogen, solidification of radioactive waste, etc.

The objectives of the laboratory buildings mentioned above are to guarantee the CEFR's safe and steady operation, to support engineering applications of the CEFR and the follow-up of fast reactor design, and to train new operators and researchers on new sodium technology. In general, the trainees are from the nuclear energy or alkali metal industry fields.

3.2. France

In France, the new objective is to build a GENERATION IV reactor prototype by 2020 named ASTRID. This decision has motivated an important and rapid increase in R&D work, orientated towards design and conceptual evaluations. Two reactors are currently being dismantled, Phenix and Superphénix. It was therefore necessary to support these activities and promote education and training initiatives. To support this requirement, the Ecole du Sodium et des Métaux Liquides, Ecole des Combustibles, both located in

FIG. 1. Trainees dismantling the pipe of a sodium and potassium alloy loop.

CEA-Cadarache, and the Institut des Sciences et Techniques Nucléaires are the key schools to support the development of SFRs.

The initial objectives of the Sodium School (Ecole du Sodium et des Métaux Liquides) were to synthesize knowledge, to share it between CEA experimental facilities' operators and consequently to support R&D activities, to train operators able to work on the SFRs Rapsodie and Phénix, to train design engineers involved in the Superphénix project and to train fire brigades. Its role has always been to adapt its training content to the changing needs of reactor operation, experimentation and design activities (see Figs 1 and 2). Trainees usually belong to French companies such as CEA, EDF, AREVA, and the Nuclear Safety and Radioprotection Institute (IRSN), or any companies involved in sodium activities irrespective of whether these are related to the nuclear industry. There are ten different sessions (from 1 to 5 days) focusing on four main subjects:

(i) Physico-chemistry of sodium coolant (physical and chemical properties, purification, corrosion, contamination, cleaning and analysis);
(ii) Sodium technology (commissioning and operation, description and operation of components, instrumentation, visualization, inspection and repair, exercises — operation and intervention on the sodium loop dedicated to education and training);
(iii) Sodium safety (specific risks: sodium–water reaction, sodium fires, safety rules, prevention, intervention, exercise on a real sodium fire);
(iv) Sodium decommissioning (specific risks, dismantling techniques, sodium treatment, sodium waste storage, decommissioning of sodium and NaK facilities).

At an early stage in its creation, the Sodium School was intended to be open to foreign countries. As an example, specific training sessions were provided for German operators (1983), Japanese operators for the first startup of the Monju reactor (1990s), or in support of the PFR and DFR decommissioning projects (United Kingdom). Specific sessions were also provided to the chemical industry, such as UOP (United States of America). More recently, the Sodium School, in association with the PHENIX plant operator, has extensively increased its openings to foreign institutes, such as trainees from the CIAE in China, ROSATOM in the Russian Federation on reactor technologies, safety and operation, and the Indira Gandhi Centre for Atomic Research (IGCAR) in India, dedicated to safety. The pedagogical approach consists of a combination of various educational means: lectures, discussions and training on sodium loops.

Since 1975, more than 5000 trainees have been trained at the Sodium School.

Within the frame of the INSTN (Institut National des Sciences et Techniques Nucléaires) (http://www-instn.cea.fr), three new sessions were successively prepared since 2007 and launched:

(i) SFR history, main options, design and operational feedback;
(ii) SFR functional analysis and design; and
(iii) SFR safety and operation.

These sessions are dedicated to the orientation of the Generation IV forum studies, including the main design options, operational feedback experience, circuit and plant operation, with emphasis on transients, safety and commissioning aspects, and finally a visit of the PHENIX reactor.

Two other training sessions also exist in the INSTN catalogue:

(i) SFR: Core physics;
(ii) SFR: Beginning with the ERANOS code system.

The duration of both sessions is one week. In addition, the INSTN develops its own nuclear engineering Masters level (or specialization) degree and a catalogue of more than 200 vocational training courses, covering general nuclear issues, but mainly dedicated to water reactors.

In addition, France has an important nuclear education and training platform organized around engineering schools, universities, research laboratories, technical schools and also nuclear companies (for internal and possibly external training) or dedicated entities, for professional training. In this context, I2EN, the International Institute for Nuclear Energy (http://www.i2en.fr), set up in 2010, is federating French entities, delivering high level curricula in nuclear engineering and science and is promoting the French offer for education and training in partner countries.

FIG. 2. Trainees working on the filling of a sodium loop (the Sodium School).

3.3. India

The second stage of the Indian nuclear power programme, piloted by IGCAR, envisages the development of expertise in SFRs and associated fuel cycle technologies. In India, Fast Breeder Test Reactor has been in successful operation for the past 27 years and a 500 MW(e) Prototype Fast Breeder Reactor is at an advanced stage of construction and commissioning. A robust R&D base encompassing all the disciplines has been established. To nurture and enhance the R&D capabilities available with various academic institutes (e.g. Indian Institute of Technology, Indian Institute of Science, etc.) across the country, collaborative research is being pursued in identified areas. Several projects with well defined expected outcomes are being pursued with active participation from the scientists from IGCAR.

Taking into consideration the enhanced role FBRs are likely to play in contributing to the nuclear power component of the nation, a need to augment skilled personnel for the critical assignments to take up challenges in the design of plant and development of equipment and processes was felt. Initiating the training school programmes at IGCAR, identifying research scholars to take up the problems in interface areas for achieving breakthroughs, formulating a streamlined methodology for training the supervisors and technicians, especially with respect to reactor operation and sodium handling, are some of the avenues that have been explored to augment the skills required for various programmes.

H.J. Bhabha envisaged the training school programme at BARC as early as 1956, in order to meet the growing demand for skilled personnel to take up the challenging assignments in the Department of Atomic Energy. The training school programme has been adapted by various units of the Department with special emphasis on specific requirements of the unit, such as R&D in mission programmes of the Department or operation, maintenance and management of plants. While the training school would cater to fulfilling the requirements of scientists and engineers, specific programmes to meet the requirement for trained technical personnel for operation of reactors and reprocessing plants have been designed.

In our endeavour to nurture young talent to meet the challenges of developing safe, economically competitive and advanced fast reactors and associated closed fuel cycle technologies, the training school for fresh engineers/science post-graduates was started in September 2006. Initially, the training programme was conducted in three disciplines, i.e. mechanical, electronics and instrumentation, and chemical engineering, with orientation towards fast reactors and closed fuel cycle technologies. Realizing the need for emphasis on challenging issues related to reactor physics, safety, processing of nuclear materials, including reprocessing, etc., two additional disciplines, i.e. nuclear reactor physics and nuclear fuel cycle chemistry were subsequently commenced. Training in materials science to take up challenges in R&D in metallurgy is also in place. The training school has no permanent teaching staff and all the teachers are actively working scientists and researchers in their chosen fields of expertise. In this way, the teachers have an opportunity to put forth their research problems in the classroom. The faculty is also drawn up from other units of the Department of Atomic Energy (other than where the school is located) and also from reputed neighbouring academic institutions. The training school programme infuses trained personnel to undertake the mission programmes of the centre and Department on a continuous basis. While the training by practicing professionals ensures that implicit and tacit knowledge is shared and cultivated, continuity of intake and training sets off a ‘chain reaction’, ensuring that trained personnel is available at all times. Moreover, security of their job to those selected for training ensures their commitment. The training school courses are periodically reviewed and updated to keep abreast of the latest advances and this ensures supply of expert personnel according to the changing nature of the requirements of the Department.

We also realized that in order to progress, challenges in R&D related to FBRs operating on advanced fuel with inherent safety and economy have to be addressed. We believe that the approach to solve most of the challenges is by practicing science based technology, which is based on a solid foundation of physics, chemistry and engineering. This approach would enable us to achieve breakthroughs and also provide for an adequate resilience in front line technologies. In order to realize our goals, we have inducted young research scholars into our vibrant R&D programmes. The strength of the research scholars in the centre has been constantly increasing from fifteen about five years ago to a present sanctioned strength of two hundred. We have identified the areas which have a direct bearing on the mission related activities of the centre that include fast reactor physics and engineering, chemistry and chemical engineering, computer science, electronics and control instrumentation, materials science and engineering and reactor safety. Research scholars will be pursuing research in interdisciplinary fields that links basic sciences with engineering, such as physics and reactor engineering, chemistry and chemical engineering, etc. This will promote the involvement of basic sciences in challenging mission programmes of the Centre and also provide opportunities for original research and potential for breakthroughs. While allotting the students, we are ensuring that the guides have the credibility of a mentor, ability and good quality of his /her own research.

A dedicated training centre with all the facilities, such as mechanical, electrical and instrumentation shops and working models exclusively for fast reactors is being commissioned to cater to the training needs of the supervisors and technicians for on the job training on equipment and facilities, and for training on radiation protection and reactor operation/maintenance.

A sodium school has been in operation, which exclusively conducts training programmes in sodium related studies and also on handling sodium. The Sodium School was conceived at IGCAR with an objective of providing practical training to young engineers/scientists who are directly related to design and development of fast breeder reactor technology and personnel from industries and collaborative institutes. As part of the training, lectures from eminent scientists from India and abroad (France) are arranged and practical demonstrations are provided in the 'labview' sodium facility.

FIG. 3. Labview sodium school.

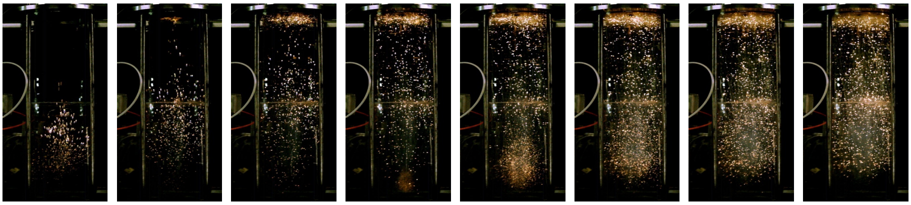

FIG. 4. Sodium spray fire scenario.

The labview sodium school (Fig. 3) is augmented with facilities to demonstrate key experiments, such as sodium spray fire (Fig. 4), sodium pool fire, sodium–water reaction, sodium–concrete interaction, sodium leak, sodium fire mitigation, etc. The Sodium School is also facilitated with literature on operating experiences over the years on all aspects of sodium. The uniqueness of the Sodium School is that, in addition to lectures, the participants will be conducting benchmark tests on various aspects of sodium.

Some academic institutes in India have initiated Masters programmes in nuclear science and technology. Several of these initiatives are being supported by IGCAR, by way of formulating the syllabus and by participation of the scientists as faculty members for conducting the courses at the host institutes. Practical training in the test reactor, engineering facilities and safety experiments is being provided to these students. Avenues for pursuing doctoral programmes for students with specialization in nuclear science and technology are made available at IGCAR.

3.4. Japan

Japan is striving to establish the FBR fuel cycle as the national energy policy, which enables uranium resources to be used more effectively, and is aiming at starting commercial operation of a demonstration FBR around 2025 in the FaCT (Fast Reactor Cycle Technology Development) project. To contribute to the support of FBRs, INITC (International Nuclear Information and Training Center) has been created and is working on human resource development using the prototype Monju FBR with a focus towards the next generation, not only for young Japanese engineers and students but also for the world, aiming at becoming a base of the international educational training. The human resource development organized by INITC is categorized into the following two tracks:

(i) The domestic educational training activity consists of FBR engineer educational training, which targets young engineers and researchers engaged in Monju, and the student educational training for enhancing the understanding of nuclear technology and increasing the interest in, and care for, environmental energy.
(ii) The international educational training activity comprises two types of nuclear technology training courses, i.e. the international sodium handling training course and the international reactor plant safety course.

Regarding the FBR engineer educational training positioned as the core of Monju education and training, the new establishment of a training framework, new construction of the Fast Reactor Training Facility in 2000 with two types of training facilities for ‘synthetically’ learning sodium handling and maintenance technologies, and remodelling the Monju Advanced Reactor Simulator, were carried out to improve the existing educational training held before the Monju sodium leak accident, which occurred in December, 1995.

The training framework was established for the FBR engineer educational training and is composed of a total of 27 training courses categorized into the following four training categories: FBR operation technical training, sodium handling technical training, maintenance technical training and FBR plant system engineering training. Each training course has been continuing even while operation of Monju has been stopped for over a decade.

With respect to the student educational training, two educational activities were proposed:

(i) The Tsuruga Summer Institute on Nuclear Energy for graduate students of the whole country, including local universities, which was organized as one of the collaborative activities with the CEA (Cadarache, France) from 2006;
(ii) The energy environmental education for high school students in the local community from 2007.

On the other hand, INITC is presently organizing two types of international educational training programmes of around one month’s duration, sponsored by the Ministry of Education, Culture, Sports, Science and Technology. One is the international sodium handling training course for China and the USA (from 2004). The other one is the international reactor safety plant course for eight countries in Asia for learning a variety of reactor safety technologies (from 2006).

In order to synthetically master the sodium handling technology, the technical subjects which should be learned were categorized as follows: sodium general knowledge, sodium physical and chemical properties, sodium loop operation techniques (sodium charge and drain operations), sodium purification control technique, sodium corrosion mechanism, treatment skill for sodium compounds, response and treatment skills against sodium piping leaks (see Fig. 5).

The following various sodium handling training systems or devices were used for educational purposes:

(i) A multi-purpose sodium training cell, (sodium firefighting training, sodium handling work training, sodium piping leak response training, etc.).
(ii) A sodium training loop: By using this training loop, both sodium charge–drain and purification operations can be covered.
(iii) A sodium combustion observation device.
(iv) A sodium property measuring device: A total of six physical properties are measured i.e. density, melting point, kinetic viscosity, specific heat, thermal conductivity and surface tension.
(v) A liquid sodium flowing observation device.
(vi) An imitation of sodium leakage pipe.

In addition, several training models particular to Monju are located in the maintenance training facility, such as sodium pump mechanical seal mock-up, fuel handling system, control rod driving mechanism and the fuel handling machine. The FBR operation technical training method was continuously improved including:

— Establishment of educational training guidelines (educational training items, frequency of educational training, training contents, teaching materials). This guideline is applied to simulator training, sodium handling and maintenance technical training.
— Introduction of simulator training evaluation manual (for individual and for shift crew).
— Introduction of e-learning system.

In addition, the Research Institute of Nuclear Engineering of the University of Fukui, established in 2009 and now located in Tsuruga, has implemented several departments, among them:

— The Department of Nuclear Engineering Science dedicated to the development of nuclear analysis codes, management of education strategy and action plans.

— The Department of Nuclear Engineering Research and Development, focusing on SFRs with a special emphasis on Monju, and also addressing LWRs, fuel technologies and their respective decommissioning.
— The department dedicated to severe accidents, i.e. to earthquake- and tsunami-proof technologies, nuclear disaster prevention and crisis management field.

With the support of the Ministry of Education, Culture, Sports, Science and Technology, this institute is aimed at developing human resources, particularly students of Fukui University, but there is also acceptance of internship from CEA–INSTN, within the frame of a bilateral exchange, and from other Asian countries.

Among the facilities used for education and training, the Monju simulator can be mentioned and also all the sodium facilities already listed.

3.5. Republic of Korea

The role of nuclear power in electricity generation in the Republic of Korea is expected to be more important in the years to come in achieving energy self-reliance. There are now 19 PWRs and 4 PHWRs in operation, 5 PWRs under construction and 4 PWRs in preparation. It had been recognized nationwide that a fast reactor system is one of the most promising nuclear options for electricity generation with an efficient utilization of uranium resources and a reduction of radioactive waste from nuclear power plants. In December 2008, the KAEC (Korea Atomic Energy Commission) authorized the R&D action plan for the Advanced SFR and the pyroprocesses to provide a consistent direction to long term R&D activities. The plan was revised in November 2011 in order

FIG. 5. Synthetic sodium leak monitoring system.

to refine the plan and to consider the available budget for SFRs with regard to the milestones, i.e. specific design of a prototype SFR by 2017, specific design approval by 2020, and construction of a prototype SFR by 2028.

The Sodium Cooled Fast Reactor Development Agency was organized on 16 May 2012 with the goal of gaining design certification for the PGSFR (Prototype Gen IV SFR). There was a phase change in the SFR development programme from key technology development in the past to overall system engineering, including SFR system (NSSS and BOP) design and optimization, integral verification and validation tests and major component development, etc. The Sodium Cooled Fast Reactor Development Agency, which has a role in funding and managing the PGSFR project, including NSSS, BOP, component design, and the development of related technology, is an affiliated organization of KAERI (Korea Atomic Energy Research Institute).

The main characteristics of the PGSFR are:

— Pool type metal fuelled reactor with power level of 150 MW(e);
— Core outlet temperature of 545^{o}C;
— Superheated Rankine cycle used for power conversion system with an optional study of supercritical CO_2 Brayton cycle as an advanced concept;
— Decay heat removal system to be selected in early of 2013 (passive decay heat removal system with 2 PDRC and 2 ADRC, RVACS type systems (reactor vessel auxiliary cooling systems).

SFR technology development efforts in the Republic of Korea commenced in June 1992 with the KAEC's approval of national mid- and long term nuclear R&D programmes. Basic research was performed until 1997. The conceptual design of the KALIMER-150 and basic technologies were developed from 1997 to 2001. The conceptual design of the KALIMER-600 was developed from 2002 to 2006. According to the nuclear technology roadmap established in 2005, an SFR was chosen as one of the most promising future types of reactor, one which could be deployable by 2030, and KAERI was developing advanced SFR design concepts to meet Gen IV technology goals from 2007 to 2011.

The metal fuel is being developed in accordance with the SFR and pyroprocess development plan. A fuel fabrication technology will be developed by 2018, and a U-Zr fuel manufacturing facility will be constructed by 2024. The U-Zr fuel will be used as a starting fuel and U-TRU-Zr fuel will replace U-Zr fuel after the verification of its in-pile performance.

On the basis of the experience gained during the development of the conceptual designs for KALIMER, KAERI has developed advanced SFR design concepts of a TRU burner that can better meet the Gen IV technology goals, and which includes three categories of activities:

(i) Advanced concept design studies;
(ii) Development of the advanced SFR technologies necessary for its commercialization;
(iii) Development of basic technologies.

A large scale sodium thermohydraulic test programme called STELLA (sodium test loop for safety simulation and assessment) is being progressed by KAERI. As the first step of the programme, the sodium component test loop, called STELLA-1, has been constructed and will start operation in 2013, representing a one year delay from the original schedule. It is to be used for demonstrating thermohydraulic performance of major components such as heat exchangers and the mechanical sodium pump and their design code's verification and validation. The second step of an integral effect test loop, called STELLA-2, will be constructed to demonstrate plant safety and to support the design approval for the prototype reactor. The overall schedule of the STELLA programme is shown in the Fig. 6.

As far as the education and training on fast reactors are concerned, there are some academic courses on fast reactors in universities, although there are as yet no fixed periodically running programmes of education and training on fast reactors at KAERI. There have been several practical courses and seminars held occasionally, when necessary, at KAERI. Several of KAERI's staff were trained in sodium schools: 3 in the French Sodium School in 1993 and 2009, and 6 in the Japanese Sodium School in 2012. There will be a strong need for well trained engineers and technicians in the Republic of Korea to be involved in the operation of the test facilities and eventually of the PGSFR. For the time being, the training which is related to the sodium technology will be done within the framework of international collaboration using the sodium loop facilities in those countries having expertise in those fields. There is the intention to build a domestic sodium training loop in the long term future.

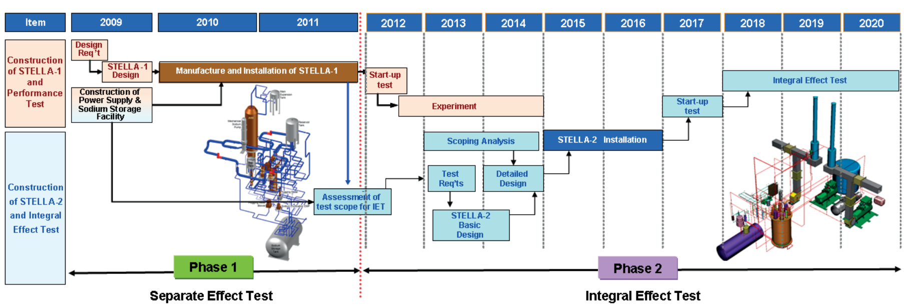

FIG. 6. Overall schedule of the STELLA programme.

3.6. United States of America

The USA has a rich heritage of fast reactor technology development, starting with the Experimental Breeder Reactor-I (EBR-I), which first generated power on the electrical grid in 1956. EBR-I was followed by the SFRs: Experimental Breeder Reactor-II, Fermi-I and the Fast Flux Test Facility. The USA was on the path to develop a commercial scale SFR when policy decisions in the 1980s changed the direction of the US fast reactor technology development away from large monolithic SFRs to modular SFRs under the Advanced Liquid Metal Reactor programme. Two modular fast reactor concepts were developed under this programme: the General Electrical Power Reactor Innovative Small Module Reactor and the Atomics International Sodium Advanced Fast Reactor designs. These advanced reactors were developed to the conceptual and advanced conceptual design stages. From approximately 1994 through 2006, there was a hiatus in fast reactor technology development in the USA. The Global Nuclear Energy Partnership programme, initiated by the USA in 2008, reinstated the domestic SFR development programme. This fast reactor R&D programme continues to under the US Department of Energy's (DOE) Advanced Reactor Concepts programme.

Under the Global Nuclear Energy Partnership and Advanced Reactor Concepts programmes, the DOE recognized the need for educating students, professors and other nuclear professionals in understanding fast reactor technology. Starting under the Global Nuclear Energy Partnership programme, a series of multi-day workshops were held with US regulatory staff that included such topics as sodium as a coolant, fast reactor materials and fast reactor safety, among others.

In addition, in August 2010, Argonne National Laboratory initiated a workshop for many university professors in the USA. The purpose of the workshop is the education and training of future generations of nuclear engineers which is acknowledged as a key challenge facing both the revitalization of nuclear energy and continuation of world class R&D in the USA. The particular topic for this workshop was fast reactor technology where the coverage is often limited or outdated in most nuclear engineering textbooks. The current R&D programmes sponsored by the DOE maintain this expertise in the national laboratories. However, the means to capture this knowledge for educational purposes is not clear. Therefore, to explore this issue, a Fast Reactor Curriculum Workshop was held on 30–31 August 2010 at Argonne National Laboratory. Since holding this very successful workshop, US national laboratory personnel have been conducting tailored workshops at various universities on fast reactor technology related topics.

The DOE also has had a separate programme to sponsor R&D at various US universities. This programme's name was recently changed from the Nuclear Energy University Program to the Consolidated Innovative Nuclear Research Program. This university programme funds nuclear energy research and equipment upgrades at US colleges and universities. This university programme plays a key role in helping the DOE accomplish its mission of leading the nation's investment in the development and exploration of advanced nuclear science and technology.

As stated in the DOE's Nuclear Energy Research and Development Roadmap, the DOE promotes nuclear power as a resource capable of meeting the nation's energy, environmental and national security needs by resolving technical and regulatory barriers through research, development and demonstration. The university programme's objectives are to support outstanding, cutting edge and innovative research at US universities by:

- Attracting the brightest students to the nuclear profession and supporting the nation's intellectual capital in nuclear engineering and relevant nuclear science, such as health physics, radiochemistry and applied nuclear physics;
- Integrating R&D at universities, national laboratories and industry to revitalize nuclear education;
- Improving university and college infrastructures for conducting R&D and educating students;
- Facilitating the transfer of knowledge from the ageing nuclear workforce to the next generation of workers.

About 20% of the DOE's nuclear energy budget is provided to support university nuclear research programmes. For sodium fast reactor technology development, the university programme has provided funding for PhD candidate students at the University of Wisconsin to develop a sodium test loop for the testing of advanced materials in sodium. For this project, one of the students developed an innovative moving magnet sodium pump and another student is working on oxygen sensor development for sodium applications.

Along with the classroom training of young engineers, the Advanced Reactor Concepts programme is also conducting R&D on various advanced technologies that support the commercialization of fast reactor technology by 2050. This fast reactor technology development includes advanced structural materials, advanced systems and components, compact reactor core configurations, innovative energy conversion systems and technologies for conducting in-service inspection. To support the technology development activities listed above, sodium facilities have been developed or are being developed to investigate sodium plugging phenomena in small channels of compact heat exchangers, the effects of sodium freeze and thaw in these small channels, the ability to drain compact heat exchangers, the testing of under-sodium viewing technology, and the testing of mechanisms and other components in a facility known as the Mechanisms Engineering Test Laboratory.

In closing, the above section describes some of the US history of SFR technology development and the ongoing efforts to ensure that there is a transfer of knowledge on advanced fast reactor technology.

3.7. IAEA

The IAEA — as an independent intergovernmental, science and technology based organization that serves as the global focal point for nuclear cooperation — promotes and supports education and training programmes for the safe, secure and efficient development of the nuclear field. The IAEA offers a wide spectrum of activities in support of education, training, human resource development and capacity building, including interregional, regional and national training courses and workshops. It also supplies training materials and training services and tools, including e-learning and on-line courses. Finally, the IAEA carries out internship programmes for the younger generation.

In particular, the Department of Nuclear Energy, often in collaboration with the Department of Nuclear Sciences and Applications and the International Centre for Theoretical Physics (ICTP) in Trieste, secures education and training in the field of fast neutron system physics, technology and applications, owing to the contributions of experts from the Member States.

Within the frame of its numerous activities in the field of fast reactors, several initiatives related to education and training have been carried out; the most recent ones being:

- School on Physics, Technology and Applications of Innovative Fast Neutron Systems, in collaboration with ICTP, Trieste, 9–20 November 2009;
- Workshop on Nuclear Reaction Data for Advanced Reactor Technologies, in collaboration with ICTP, Trieste, 3–7 May 2010;
- Workshop on Codes and Standards for Sodium Fast Reactors, Beijing, 6–8 July 2010;
- Education and Training Seminar/Workshop on Sodium-cooled Fast Reactors Science and Technology, San Carlos de Bariloche, Argentina, February 2011;
- Workshop on Environmental Degradation of Components in Nuclear Power Reactors (including fast neutron systems), in collaboration with ICTP, Trieste, 5–16 March 2012;

— Education and Training Seminar/Workshop on Fast Reactor Science and Technology, San Carlos de Bariloche, Argentina, 1–5 October 2012 (see Fig. 7).

As a biennial education and training course in the field, the IAEA is organizing a School on Physics, Technology and Applications of Innovative Fast Neutron Systems and Related Fuel Cycles, which will be held at ICTP, Trieste in September 2013.

3.8. European Union

In Europe, the Strategic Research Agenda of the Sustainable Nuclear Energy Technology Platform (grouping over 100 stakeholders from industry and research organizations) has selected three fast neutron reactor systems as a key structure in the deployment of sustainable nuclear fission energy. Fast reactor development needs an important technology support to finalize their innovative design and to assess their safety. The SFR concept is currently considered as the reference within the European strategy and ASTRID as the demonstrator. Europe also supports developments towards lead cooled fast reactor MYRRHA, Accelerator Driven System ALFRED projects and the gas cooled fast reactor ALLEGRO. Several education and training initiatives are organized with the support of the European Commission to the European Nuclear Education Network, and within the frame of projects co-funded through the Euratom Framework Programme.

The European Nuclear Education Network Association (http://www.enen-assoc.org) was established in 2003 to preserve and further develop expertise in the nuclear fields through higher education and training.

FIG. 7. IAEA meeting in San Carlos de Bariloche, Argentina, October 2012.

The European Nuclear Education Network Association currently has over 60 members, mainly in Europe. This objective is realized through the cooperation between universities, research organizations, regulatory bodies, industry and any other organizations involved in the application of nuclear science and radiation protection. The European Nuclear Education Network Association fosters student mobility in Europe and beyond.

Within the frame of the European Commission's FP7 project CP-ESFR (Euratom), since 2009, 5 European Sessions (see Fig. 8) dedicated to SFRs have been organized by the Ecole du Sodium et des Métaux Liquides at CEA-Cadarache in France, University of 'La Sapienza' in Italy, the Karlsruhe Institute of Technology in Germany and the University of Madrid (Spain). More than 120 trainees and PhD students were welcomed during these five sessions. Within the frame of a new project, ESNII+, a major effort dedicated to fast neutron reactors cooled by sodium, lead and gas is foreseen. Eight seminars and two summer schools have been defined and will be organized between 2014 and 2017, dedicated to various topics such as:

— Fuel properties and fuel transient tests;
— Core neutronic safety issues;
— Instrumentation for fast neutron reactors;
— Thermohydraulics and thermomechanical issues;
— Mitigation of seismic risks;
— Coolant physicochemistry and dosimetry, and quality control strategy;
— Safety assessment of fast neutron reactors;
— Severe accidents in fast neutron reactors;
— Sitting and licensing of fast neutron reactors.

FIG. 8. Trainees attending a European session (Cadarache, November 2010, FP7 Project CP-ESFR).

4. CONCLUSION

The result of this ambitious and long term strategy is first a share of knowledge gained through experimental studies carried out in research laboratories as well as feedback from fast reactor operation, secondly, standardized information on safety and finally, the creation of a SFR community is promoted, able to debate, share knowledge and suggest new tracks for better definition of design and operating rules.

These education and training strategies and initiatives are constantly supported by schools, seminars and workshops. They are key elements for the design and operation of SFRs, and more particularly to support the development of new projects, the safe operation of existing reactors and the creation of a new generation of skilled nuclear engineers in the field.

National and international organizations currently undertaking education and training activities in the field of SFRs and presented in this paper are also very keen to collaborate and to share their own experience, thanks to common initiatives and the invitation of foreign teachers, as well as to launch common initiatives for the highest benefit of the entire SFR community.

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OVERVIEW OF US KNOWLEDGE PRESERVATION PROGRAMME FOR FAST FLUX TEST FACILITY DATA

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Abstract

One of the goals of the US Department of Energy's Office of Nuclear Energy (DOE-NE) is to preserve the knowledge that has been gained in the United States of America on liquid metal reactors (LMRs). In addition, preserving LMR information and knowledge is also being pursued as part of a larger international collaborative activity conducted under the auspices of the IAEA. Specific knowledge preservation activities were initiated by DOE-NE under the Fuel Cycle Research and Development programme and are continuing under the Advanced Reactor Concepts programme. The primary objective of these activities is to collect, organize and preserve technical information that could support the development of an environmentally and economically sound nuclear fuel cycle. The DOE programme includes activities to preserve information from the two most recent LMRs constructed and operated within the USA: the Experimental Breeder Reactor II (EBR-II) and the Fast Flux Test Facility (FFTF). The FFTF is the most recent LMR to operate in the USA (1982–1992) and was designed as a fully instrumented test reactor with on-line, real time test control and performance monitoring of components and tests installed in the reactor. The 10 years of operation of the FFTF provide a very useful framework for testing the advances in LMR safety technology based on passive safety features that may be of increased importance to new designs after the events at Fukushima. This report describes the knowledge preservation activities related to FFTF legacy information including data from the design, construction and startup of the reactor, but more importantly is focused on preserving information obtained from 10 years of successful operations, such as data from the extensive irradiation testing of fuels and materials which was largely unpublished. In order to ensure protection of information at risk, the programme to date has sequestered reports, files, tapes and drawings to allow for secure retrieval. The FFTF knowledge management programme includes a disciplined and orderly approach to respond to client's requests for documents and data in order to minimize the search effort and ensure that future requests for this information can be readily accommodated. This report summarizes the current status and accomplishments of the FFTF knowledge preservation activities and provides insight into the lessons learned that may benefit other knowledge preservation activities.

1. INTRODUCTION

The Fast Flux Test Facility (FFTF) is the most recent liquid metal reactor (LMR) to be designed, constructed and operated by the US Department of Energy (DOE). The FFTF operated from 1982 to 1992. The technologies employed in designing and constructing this reactor, along with information obtained from tests conducted during its operation, are currently being secured and archived by the Department of Energy's Office of Nuclear Energy (DOE-NE). Project efforts to retrieve and preserve critical information related to the FFTF have been periodically updated and presented in scientific and technical forums [1–6] . The engineering knowledge from the design, construction and operation of the FFTF represents a huge investment and cannot be duplicated.

Knowledge preservation at the FFTF is focused on the areas of design, construction, startup and operation of the reactor. The primary function of the FFTF was to be a test reactor. Therefore, the focus is to preserve information obtained from the irradiation testing of fuels and materials performed in the FFTF. In order to ensure protection of information at risk largely because of ageing/degrading storage media and no centralized document repository, the programme to date has focused on sequestering and secure retrieval of FFTF records.

Located on the Hanford site in Washington State, the FFTF reactor plant is one of the facilities intended for decontamination and decommissioning consistent with the cleanup mission on this site. The reactor facility has been deactivated and is being maintained in a 'cold and dark' minimal surveillance and maintenance mode until final decommissioning is pursued.

2. FFTF DESCRIPTION

A picture of the FFTF plant and its location at the Hanford site in Washington State is shown in Fig. 1. Figure 2 provides a diagram of the FFTF reactor plant and key parameters are listed in Table 1. A cutaway of the reactor is shown in Fig. 3. Since it was designed as a flexible test reactor, the FFTF did not have steam generators but included dump heat exchangers. It was designed to provide a prototypical test bed with respect to temperature, neutron flux level and gamma ray spectra for fast reactor fuels and materials testing. The FFTF was designed as the most extensively instrumented fast spectrum test reactor in the world, with proximity instrumentation of temperature and flow rate for each core component as well as contact instrumentation and gas and electrical connections for special test positions. Figure 4 shows an example FFTF instrumented test assembly.

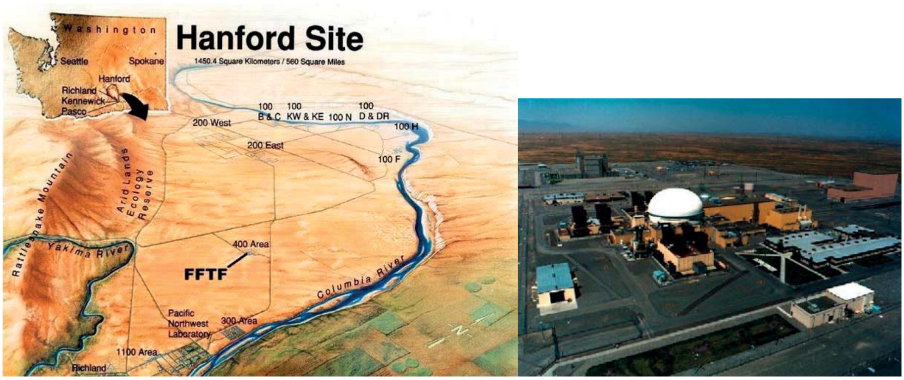

FIG. 1. FFTF at the Hanford site.

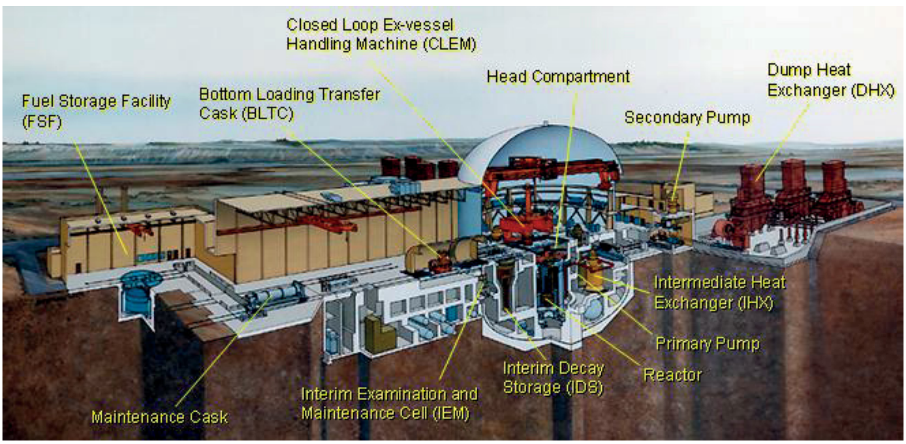

FIG. 2. FFTF reactor plant.

TABLE 1. FFTF PARAMETERS

Parameter	Value
Thermal power	400 MW
Coolant	Sodium
Coolant inlet/outlet temperatures	360/526°C
Coolant loops	3

TABLE 1. FFTF PARAMETERS (cont.)

Parameter	Value
Driver fuel material	$(Pu\text{-}U)O_2$
Enrichment zones	2
Core height	91.4 cm
Core diameter	120 cm
In core driver, test locations	82
Instrumented through head	8

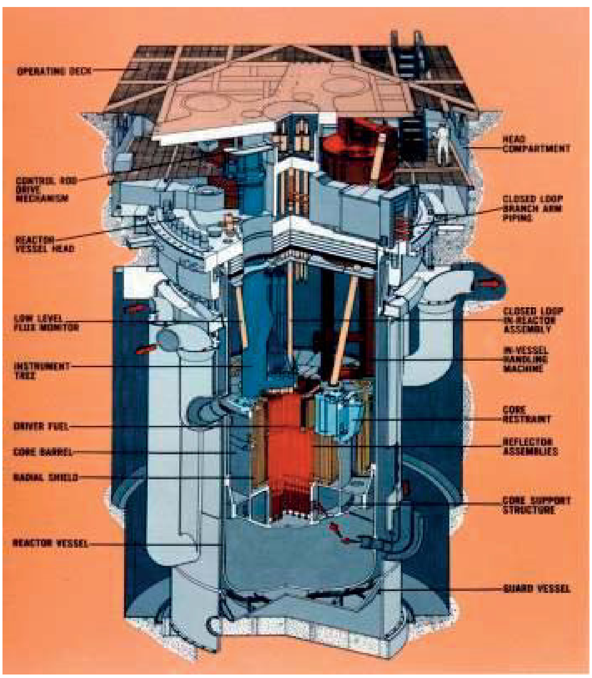

FIG. 3. FFTF reactor.

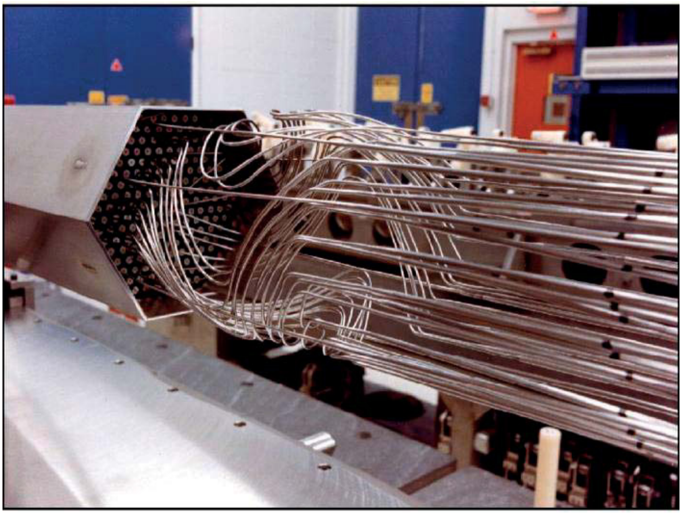

FIG. 4. Instrumented FFTF test.

Special FFTF data measurements feature include:

- — Primary and secondary loop hot and cold leg temperatures and flow rates, neutron detectors, pump speed indicators;
- — Assembly outlet temperatures for each core location with a response time of minutes;
- — Fast response thermocouples for assembly outlet temperatures for two core locations with a response time of seconds;
- — Two fuel tests with high response wire wrap thermocouples on fuel pins during the natural circulation tests at startup;
- — The plant data system (PDS) recorded 1300 variables at 1–60 second intervals;
- — The experimenter's data system (EDS) recorded several hundred selected parameters at up to 0.1 second intervals.

FFTF test data fall into the following categories:

- — Startup testing: The Acceptance Test Program documented the design and startup process for the reactor. The Reactor Characterization Program provided detailed neutron and gamma ray characterization of the in-core and ex-core environments.

— Passive safety testing: The extensive instrumentation and characterization of the reactor and heat transport system supported a wide variety of tests performed to demonstrate the safety characteristics of LMRs. The FFTF provided important operational data on the performance of liquid sodium as a heat transport medium and demonstrated the reliability and efficiency of pumps, valves and other vital components for more than 20 years. During operation, the passive safety test programme included steady state and dynamic measurements of reactivity feedback with changes in power, coolant flow rate and coolant temperatures.
— Plant data: Detailed plant data acquired during operation, such as assembly outlet temperatures and flow rates, coolant system temperatures and flow rates, and reactor vessel temperatures, were recorded on magnetic tapes by the PDS or EDS. Operational and test data (~1500 variables) at sub-second frequencies were routinely recorded on magnetic tape by these data acquisition systems.
— Fuels and materials irradiation testing: Irradiation tests were successfully conducted for a wide variety of test assemblies, such as advanced fuels (MOX, metal, carbide, nitride), blankets, control and shim absorbers, cladding and duct materials, structural materials, reflectors and spectral tailoring assemblies for special tests.

Each of these types of data and how they were generated and stored are described in the following sections.

3. STARTUP TESTING

3.1. Startup process

The FFTF underwent a systematic, rigorous and comprehensive startup of each plant system to verify that the design, documentation, installation and operation conformed to the design and safety requirements specified in the system design documents and the final safety analysis report. Formal testing began in 1978, but some preliminary testing was conducted as early as 1974. The startup test programme was officially completed in 1982 with 166 tests performed. The architect-engineer and prime constructor of FFTF was Bechtel Corporation, which was also the design contractor for many of the plant’s auxiliary systems. The main design contractor for the reactor was the Westinghouse Advanced Reactor Division with many of the reactor support systems designed by Atomics International and Aerojet Manufacturing Company. When the Atomics International and Aerojet Manufacturing Company designs were completed, the

Hanford Engineering Development Laboratory assumed responsibility for their designs through the construction and startup phases. The overall startup activities were controlled by the Westinghouse Hanford Company, which managed the Hanford Engineering Development Laboratory for the DOE. The DOE project control of FFTF was managed through a local project office.

The startup testing process consisted of three types of test: construction tests, pre-turnover engineering tests and acceptance tests. The first two types of tests were conducted prior to formal turnover of a plant system from Bechtel to the Hanford Engineering Development Laboratory and the acceptance tests were then performed after turnover. As the timing of system turnovers varied, it was not uncommon for all three types of tests to run concurrently during the startup period. The main document for control of the FFTF startup testing was the FFTF startup test plan, which describes the administrative procedures used and the general responsibilities of the various organizations involved. Construction testing was conducted by Bechtel on all portions of the FFTF to ensure that construction was completed in accordance with the drawings and specifications. Pre-turnover engineering tests had to be performed at a particular step in the construction sequence before further assembly made later testing and correction of problems impractical or impossible. Turnover was the transfer of custody (responsibility for operation, maintenance and safety) of a portion of the plant from the construction contractor (Bechtel) to the operating contractor (Hanford Engineering Development Laboratory). The startup testing documents, including QA records, have been identified and preserved.

3.2. Acceptance testing

The acceptance testing programme was conducted by Hanford Engineering Development Laboratory personnel following completion of construction testing and turnover to provide confirmation of design, construction and functional performance of the FFTF. Acceptance testing was divided into five phases: (1) Pre-operational tests, (2) system startup tests, (3) hot functional tests, (4) nuclear startup tests and (5) power ascension tests. Each of the five phases included the following documents: (1) test resume (used for test planning and includes a summary of test objectives, plant status required, and any special test equipment required), (2) test specification, (3) test procedure, (4) test operating procedures, (5) calibration procedures, (6) data report and (7) evaluation report.

3.3. Initial physics tests

The initial physics testing during the acceptance testing programme provided the first confirmation of the predictions and prediction techniques developed during the design process. The initial fuel loading was carried out by trisector (1/3rd of the core) to accommodate the special fuel handling equipment in the FFTF. Data preserved from the initial critical configuration and subsequent full-core critical configuration with fresh fuel prior to any power operation could be developed into LMR physics benchmarks. These data would be invaluable for use as experimental benchmarks in the development of reactor physics/kinetics codes and models. Subcritical reactivity effects were assessed with the modified source multiplication technique that was calibrated with an inverse kinetics analysis of a rod drop experiment. Two different dynamic testing methods confirmed the basic reactivity feedback model of the FFTF and its wide margin to instability. The first method consisted of scramming a nearly fully inserted rod to initiate a power transient. This ‘rod drop’ technique is similar to that used at EBR-II for many years. The second method, multi-frequency binary sequence, moved a control rod in small, programmed steps about a mean rod position. The reactivity feedback parameter was measured as a function of the driving signal frequency. The agreement between the two methods over the range of frequencies important for FFTF stability evaluations was excellent, especially considering the significant differences in the experimental techniques. The FFTF experience with these and other operational physics tests can be found in the archived reports and extracted from the plant data [7]. Initial physics test reports have been identified and archived.

3.4. Reactor characterization testing

The primary purpose of the reactor characterization programme was to ensure that the test conditions supplied to the FFTF irradiation experimenters were accurate. It also provided data at the high temperatures encountered in operating LMRs that could be used to adjust the calculational tools used at the FFTF and future LMRs.

Prior to full power operation, zero power testing was conducted in a special ‘in-reactor thimble’, a special central test assembly with access through the reactor head that provided a controlled environment at ~10°C near the core center for testing. Measurements included both passive and active neutron and gamma detectors, including:

— Absolute fission chambers (^{232}Th, ^{233}U, ^{235}U, ^{238}U, ^{237}Np, ^{239}Pu, ^{240}Pu ^{241}Pu);
— Proton recoil proportional counters;
— Nuclear research emulsions;
— Traversable fission chambers (^{232}Th, ^{233}U, ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, ^{241}Pu);
— Neutron dosimetry, including short lived reaction products ^{23}Na (n,γ)^{24}Na, ^{41}K(n,p)^{41}Ar, ^{81}Br(n, γ)^{82}Br;

— Gamma ray calorimeters, ionization chambers, Compton recoil spectrometers, thermoluminescent dosimeters.

Special core and reflector assemblies containing approximately 2200 dosimeters up to ±150 cm from the core midplane to characterize the neutron flux and reaction rate environment were irradiated at full power in the first 8.6 effective full power days of operation during the startup core characterization tests of the acceptance testing programme. Burnup measurements were also made on special removable fuel pins irradiated during this test. Absolute fission rate measurements confirmed the accuracy of the thermohydraulic power calibration instruments and methods. The data provided detailed neutron spectrum information and spatial reaction rate detail. One of the most significant uses of the data was to validate the cross-sections used in the FFTF reload designs and could be used for the validation of similar data in the development of new codes and models. The FFTF experience with these physics tests was preserved in the archived reports [7].

4. PASSIVE SAFETY TESTING

4.1. Incentives for incorporating passive safety features in LMR designs

Accidents at Unit 4 of the Chernobyl station and Unit 2 at Three Mile Island changed the safety paradigm of the nuclear power industry. New emphasis was placed on assured safety based on intrinsic plant characteristics that protect not only the public, but the significant investment in the plant as well. Such plants can be considered to be ‘passively safe’ since no active sensor/alarm system or human intervention is required to bring the reactor to a safe shutdown condition. The LMR has several key characteristics needed for a passively safe reactor: reactor coolant with superior heat transfer capability and very high boiling point, low system pressures and reliable negative reactivity feedback. The credibility of a passively safe LMR design rests on the validity of analytical methods used to predict passive safety performance and the availability of relevant test data to calibrate design tools. Passive safety design requires refined analysis methods for transient events because treatment of the detailed reactivity feedbacks is important. Similarly, analytical tools should be calibrated against actual test experience in existing LMR facilities. The FFTF was intentionally designed to be the most highly instrumented test reactor ever built. Data monitoring capabilities included in-vessel and ex-vessel neutron flux, coolant outlet temperature and flow for every core location, as well as heat transport system temperatures and flows. In addition, eight core locations allowed extensive contact instrumentation for tests. Fast response thermocouples provided an unprecedented level of detail.

4.2. Passive safety testing at the FFTF

During operation, a passive safety test programme was conducted, consisting of steady state and dynamic measurements of reactivity feedback with changes in power, coolant flow rate and coolant temperatures [1]:

— Steady state reactivity feedback tests were conducted to separate component reactivity feedbacks. Different reactivity feedbacks were emphasized in 198 separate measurements, separating fuel temperature and structural temperature effects. Reactor power was varied from 10% to 100%, coolant flow rate was varied from 67% to 100%, and coolant inlet temperature was varied from 303°C to 369°C.
— Delayed pony motor trip test was the first test to verify that the reactor would transition to natural circulation from nearly isothermal refuelling conditions with the primary pumps driven by pony motors without experiencing excessive core temperatures and also demonstrate the performance of fast response thermocouples in two assemblies.
— Steady state natural circulation tests demonstrated natural circulation performance.
— Controlled flow transient test by decreasing the flow rate with no control rod movement confirmed dynamic reactivity feedback models under loss of flow conditions.
— Loss of flow without scram tests with gas expansion modules (passive shutdown devices) demonstrated transition to pony motor flow and then to natural circulation flow from up to 50% power and 100% flow.
— An inadvertent pump start with a gas expansion module test was demonstrated to address reactivity insertion concerns.

The principal objectives of the FFTF passive safety test programme were to: (1) verify natural circulation as a reliable means to safely remove decay heat, (2) extend passive safety experience to a large size LMR and obtain data for validating design analysis computer codes and (3) develop and test passive safety enhancements that might be used for future LMRs.

Prior to startup, the US NRC reviewed the FFTF final safety analysis report but required tests to demonstrate the transition to natural convection circulation. These tests were performed at startup in 1980. With the reactor at 100% power and flow, the pumps were turned off and the control rods were scrammed. Special instrumented fuel open test assemblies were used to provide direct real time measurements of temperatures of individual fuel pins at several axial levels to verify the natural circulation decay heat removal.

Passive safety tests conducted during 1986 included static experiments to measure the reactivity feedback effects in off-normal conditions, controlled transient experiments to measure feedback time constants, and prototypical transient experiments to demonstrate passive safety margins. A loss of flow without scram test was made from 50% power with nine gas expansion modules, a passive safety enhancement assembly developed at the FFTF that increases neutron leakage (and decreases core reactivity) when pump flow is stopped. In 1987, the FFTF completed additional passive safety tests, including anticipated transients without scram tests. These tests were designed to provide data sufficient to allow separation of fuel temperature effects from structural temperature effects and were instrumental in improving understanding of reactivity feedback mechanisms and demonstrating passive safety margins in LMRs. The FFTF experience with all of these passive safety tests can be found in the archived reports and further details can be extracted from the archived plant data, as described in the next section [7].

4.3. Plant data

Detailed plant data acquired during these passive safety tests, such as assembly outlet temperatures and flow rates, coolant system temperatures and flow rates, and reactor vessel temperatures, were recorded on magnetic tapes by the normal PDS or EDS. During plant operation and testing periods, operational and test data were routinely recorded on magnetic tape by these data acquisition systems. The PDS recorded normal plant parameters (over 1300 variables) at frequencies up to once per second. The EDS recorded key parameters that were a subset of PDS recorded parameters, data from instrumented tests in the reactor, plus several reactor parameters used in experiment analysis that were not recorded by the PDS. Recording frequencies on the EDS were as high as once every 0.1 s, but for the passive safety and related tests, response times did not warrant recording frequencies higher than once per second. The number of parameters recorded by the EDS varied depending on how many instrumented tests were in the reactor. With no instrumented tests, the number of EDS recorded parameters was normally 100–120.

In 2009, the FFTF passive safety test plant data were prioritized for retrieval and processing to ensure that it would be available for future use. During passive safety tests conducted during FFTF cycles 7A, 7C, 8B and 8C power operations, and during cycle 12B-1 refuelling outage, 105 PDS and 15 EDS tapes were recorded. These tapes were recovered, copied and converted to ASCII text files. One text file was created for each PDS or EDS recorded tape.

Over 100 documents relevant to passive safety tests were recovered, scanned and catalogued [1] and all passive safety and related tests, and the time periods over which they were conducted, were identified. The identifiers for the data tapes covering the associated time periods were obtained from a log book retrieved from archival storage. Many of the archival storage locations of the PDS and EDS tapes were identified using keyword searches of the records centre index. Unfortunately, keywords entered for the data tapes as they were placed into archival storage were not consistent over time, making it difficult to locate some of the tapes. Documentation and software were recovered for the system used for processing tapes into a centralized database, managing the database and retrieving the stored data. Pieces of this software were modified to read and interpret the data from the tapes. The passive safety test data has been successfully located, retrieved, extracted and preserved on modern media. A web based FFTF passive safety test database is being created for accessing these data.

5. ACCOMPLISHMENTS

The future accessibility of information from the design, construction and operation of the FFTF was in doubt due to media deterioration and the lack of keyword linkage to previous programmes. In order to ensure protection of information at risk, the programme has focused on sequestering unsecured reports, files, tapes and drawings to prevent loss. Retrieval and processing of information has been selectively based on current DOE-NE programme interests and can be made available to authorized users within the DOE programme. Mechanisms could also be developed for sharing this information through international exchange agreements. Examples of specific accomplishments include:

- More than 400 boxes of FFTF information, several hundred microfilm reels including Clinch River Breeder Reactor information, and 40 boxes of information on the Fuels and Materials Examination Facility were secured as the FFTF buildings were being cleared.
- Extensive documentation of FFTF design standards, specifications, procedures and operating experience has been preserved and is retrievable. Examples include technical specifications, control room operating procedures, reactor development and technology standards, Hanford Engineering Development Laboratory standards, equipment and component procurement specifications, startup reactor characterization reports, and chief operator control room log books.

— A process for locating and retrieving PDS and EDS tapes in records storage (over 4000 binary tapes generated during plant operations, some more than 30 years old) was developed and applied successfully to retrieve and preserve data from the FFTF passive safety test programme. The passive safety test data have been successfully located, retrieved, extracted and preserved on modern media. A major uncertainty was the condition of the 25 year old tapes. To evaluate the tape condition, several non-essential tapes from the same time period were obtained and visually examined and tested. A 9 track tape drive package that works on a standard PC was obtained. The software included with the package can be used to transfer tape contents to disk. All test tapes were copied successfully to a standard PC using the new equipment. Therefore, the decision was made to read tapes containing passive safety test data in-house rather than sending the tapes to an off-site specialist in recovery of data from magnetic media. To date 120 passive safety test tapes have been successfully read and the binary data recorded to disk with minimal problems. The binary plant data were decoded and converted to ASCII format for further processing [2].
— An FFTF passive safety test database is being created and is accessed through a series of web browser HTML pages generated using Perl scripts that invoke adapted FFTF Fortran programs to produce user specified data displays.
— The fast reactor fuels testing library contains information related to fuel irradiation testing: the core demonstration experiment, driver fuel evaluation programme, fuel cladding and duct irradiation swelling characteristics, high burnup metal and MOX fuel tests, cladding and duct tests, disassembly records, dimensional profilometry, gamma spectrometry, neutron radiography, fission gas analysis, metallography, photomicrographs and procurement records. This information has been secured and data packages prepared for topics of interest to the DOE.
— Fuels, neutronics, structural and thermohydraulic analysis codes, including correlations from actual test data used to interpret test data and design fuel have been archived.

6. DATA PRESERVATION

Information from the design, construction and operation of the FFTF was at serious risk of being irretrievably lost as the facilities associated with the reactor were being shut down. Reports, drawings and data tapes were rescued as the facility was being deactivated. A large quantity of information had been stored on several different systems at the Hanford site during the design, construction and operation of the FFTF over a period of almost 20 years. Approximately 600 000 FFTF related engineering documents and correspondence are stored in the historical site records system. The Fuels and Materials Library contains over 1155 boxes of information, which translates into ~640 m (2100 vertical feet) of documents, or ~6.3 million pages. Many of these documents have no electronic counterpart and are difficult to fully digitize. In addition, nearly 800 boxes of records were transported from the FFTF 400 area QA vault to records holding storage as the FFTF was closed. The test results information exists in several different formats, depending upon the final stage of the test evaluation. Capture of tacit knowledge is necessary to preserve the full value of this information. The collected and stored documentation is more than what is available from other sources. For example, it includes complete sets of drawings of the reactor plant, operations manuals, training manuals, system design descriptions, chief operator logs, operations and maintenance manuals, cycle and outage reports, and procurement specifications. As documents and data from these systems are successfully retrieved to meet data requests and programme milestones, they are being organized and stored in an electronic database. A disciplined and orderly approach has been developed to respond to client's requests for documents and data in order to minimize the search effort and ensure that future requests for this information can be readily accommodated.

Knowledge management activities include ingesting documents into the PNNL Total Records Information Management document management system, which provides 'on demand' document identification and prioritization, full test indexing of scanned/'OCRd' PDF files, searchable metadata fields, and simple browse and report capabilities. Prioritization of documents to be ingested is driven mostly by external document requests. Archiving of FFTF data, including both the reactor plant and the fuel test information, is being performed in coordination with other data archiving efforts under way under the aegis of the advanced reactor concepts and fuel cycle research and development programmes. All information is being stored and categorized, consistent with the IAEA international standardized taxonomy, and is being converted to electronic format compatible with a general search engine being developed by INL.

7. LESSONS LEARNED

Some of the lessons learned from efforts to locate, extract and preserve FFTF data include:

— Documentation of the rigorous and successful testing programme at the FFTF was thorough and comprehensive, with official records routinely archived.
— Records storage is only useful if the records can be located. A systematic and consistent method of storing non-paper records, such as sequential or special box numbers, would have greatly increased the efficiency of locating the boxes containing the data tapes.
— Information critical to interpreting the raw data must be preserved along with the data.
— Difficulties were experienced with a few of the plant data tapes, and paths are being considered to deal with these problematic tapes.
— The majority of tapes that were successfully read contained a few blocks of data that could not be read. The data loss frequency experienced is acceptable because important parameters are recorded at higher rates than necessary to analyse the passive safety tests.

8. CONCLUSIONS

The future accessibility of information from the design, construction and operation of the FFTF has been substantially increased by the development and application of a knowledge management programme and methods for locating, retrieving and processing the historical information. The data from the FFTF startup tests provide a roadmap for a disciplined, organized approach that will be very useful for planning the startup of new LMRs. The ten years of successful operation of the FFTF provided a very useful framework for testing the advances in LMR safety technology based on passive safety features that may be of increased importance to new designs. The FFTF information provides realistic design specifications and experimental results that will be very useful to innovative designers seeking to optimize the design of new LMRs. The USA is emphasizing large scale computer simulation and modelling. The FFTF reactor characterization programme data and passive safety testing data provide the basis for creating benchmarks for validating and testing coupled thermal hydraulic/neutronic/mechanical codes. These could be especially important for LMR beyond design basis accidents and severe accidents. Mechanisms could be developed for sharing this valuable information through international exchange with other LMR programmes around the world. An indication of the value of this information is given by the fact that this information is at a level of detail and depth sufficient to rebuild the reactor plant, or alternatively, sufficient to design, construct and build a similar, although not identical, reactor.

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ENABLING FACTORS OF KNOWLEDGE MANAGEMENT: A STUDY IN THE CONTEXT OF A FAST REACTOR RESEARCH AND DEVELOPMENT ORGANIZATION

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Abstract

Knowledge management is an integrated and systematic approach to identifying, managing, archiving and sharing an organization's knowledge collectively in order to help and achieve the mission and vision of the organization. Nuclear reactor technology is a complex field as it involves many disciplines of science and engineering. Knowledge is a strategic asset for any organization, industry or plant and knowledge management is very vital for its survival and growth. Knowledge management in the context of a nuclear power plant acquires a much more important role, because of the long timescales involved, high technological excellence required, stringent safety regulations, and difficulties in attracting and retaining a talented workforce, etc. Added to this, there is always a challenge to improve safety and reduce the unit energy costs. Vast amounts of nuclear knowledge have been developed and accumulated through the decades of R&D and operational experience. This knowledge is of paramount importance for the continued use of existing nuclear installations and future innovations. Recognizing the importance of nuclear knowledge management, many nuclear organizations have initiated formal knowledge management practices. However, these organizations are yet to achieve higher levels of maturity in their knowledge management practices. In order to achieve higher levels of maturity, the organizations need to identify the enabling factors and ensure availability. This paper discusses these enabling factors identified in a fast reactor research and development organization, through a survey.

1. INTRODUCTION

Knowledge management is an integrated and systematic approach to identifying, managing, archiving and sharing an organization's knowledge collectively in order to help and achieve the mission and vision of the organization. Knowledge management in the context of a nuclear power plant acquires a much more important role because of the long timescales involved, high technological excellence required, stringent safety regulations, and difficulties in attracting and retaining a talented workforce, etc. Nuclear knowledge is knowledge specific or relevant to nuclear related activities including (but not limited to) technical engineering knowledge (Yanev, 2009) [1]. There are many stakeholders for nuclear knowledge such as designers, consultants, operators, vendors, academic and R&D institutions, governments, regulators, international organizations, etc. Nuclear knowledge has been accumulated over many decades with specific experience in predesign, design, construction, safety analysis, commissioning, operation and maintenance and R&D.

This knowledge is of paramount importance for the continued use of existing nuclear installations and future innovations. Effective mechanisms need to be developed for preservation of this knowledge and its transfer to successive generations. Recognizing the importance of nuclear knowledge management, many nuclear organizations have initiated formal knowledge management practices. However, many of these organizations are yet to achieve higher levels of maturity in their knowledge management practices. Implementation of knowledge management in an organization involves significant changes in the organizational processes. Several studies have proposed many key variables for successful implementation of knowledge management, which are called critical success factors. Alternatively, these factors are also called enabling factors. In the negative direction, these factors can be called inhibiting factors. For example, for the enabling factor ‘availability of user friendly technology infrastructure’, the corresponding inhibiting factor is ‘lack of user friendly technology infrastructure’. Rao (2003) identified a knowledge management framework with ‘8Cs’, i.e. connectivity, content, community, culture, capacity, cooperation, commerce and capital [2]. The study of Chong et al. (2005) has identified eleven critical factors for successful implementation of knowledge management [3]. McCabe (2003) identified ten critical success factors to establish knowledge management as an enterprise wide discipline [4]. Holthouse (2003) identified ten critical success factors of knowledge management [5]. Rao (2003) discusses the common mistakes made when implementing knowledge management and their possible solution. Fahey and Prusak(1998) identified eleven ‘sins’ of knowledge management and recommended their possible solution [6]. After studying thirty-one knowledge management projects in twenty-four companies, Davenport et al. (1998) identified eight success factors [7]. Many of these factors are related to organizational culture, organizational processes, leadership, technology infrastructure, etc. In order to achieve higher levels of maturity, the organization needs to identify the enabling factors pertaining to it and ensure their presence. Alternatively, the organization can identify the inhibiting factors and eliminate them. This paper discusses the inhibiting factors identified in a fast reactor research and development organization, through a survey. The inhibiting factors were selected instead of enabling factors because of the ease of collecting the responses from participants. Also, it is widely followed in the literature (Bukowitz and Williams, 1999) [8].

The remaining part of this paper is organized as follows. Section 2 details the organizational context, Section 3 discusses the study, Section 4 deals with the analysis, Section 5 describes the participant profile and Section 6 is the conclusion.

2. ORGANIZATIONAL CONTEXT

This study is carried out in the context of a fast reactor R&D organization. The complexity of managing knowledge in such an organization involves the twin complexity of nuclear knowledge management and R&D knowledge management. The organization initiated formal knowledge management practices a few years earlier. It developed and documented a formal knowledge management policy for the organization. It implemented an interconnected technology infrastructure for knowledge preservation and sharing, with freedom for individual groups to organize their own knowledge repository, which is the knowledge management portal of the organization. It also created part-time knowledge management roles such as Chief Knowledge Officer and Group Knowledge Officer, with a task force constituted by the Director, who is the Chief Executive Officer of the organization. In the next section, the activities of the organization and its groups are briefly discussed.

2.1. Groups and activities

The main organizational activities are R&D with respect to fast reactors. The organization has eleven major technical groups and two non-technical groups. Only ten technical groups are considered for the study. Though the organization is predominantly R&D oriented, it has groups which are carrying out technical services, operation and maintenance and project execution. A brief description of the activities of the groups is listed in Table 1 (IGCAR, 2011) [9]. The actual names of the groups are not mentioned for reasons of confidentiality.

TABLE 1. GROUPS AND ACTIVITIES

Group	Activities
G1	Carrying out R&D and providing analytical support with respect to all the chemistry aspects of the organization
G2	Developing electronics and instrumentation systems, providing computational and data communication services to the organization
G3	Providing engineering services to the organization
G4	Design and project execution
G5	Development and testing of models and prototype components
G6	Carrying out basic research and applied research
G7	Carrying out basic research
G8	Carrying out design and R&D
G9	Plant operation and maintenance
G10	Technology development, R&D activities and project execution

The organizational web site (www.igcar.gov.in) states the following facts:

- The organization was established in 1971 with a clear mission to conduct broad based multi-disciplinary programmes of scientific research and advanced engineering, directed towards the development of sodium cooled fast breeder reactor technology.
- Over the years, the organization has established comprehensive R&D facilities covering the entire spectrum of fast breeder reactor technology related to sodium technology, reactor engineering, reactor physics, metallurgy and materials, chemistry of fuels and its materials, fuel reprocessing, reactor safety, control and instrumentation, computer applications, etc., and the organization has developed a strong base in a variety of disciplines related to this advanced technology. Apart from thrust areas related to nuclear technology, the organization has credentials as a leader of basic research in various ‘frontier’ and topical areas.

- The organization has staff strength of 2816 including 1274 engineers and scientists. The annual outlay of the organization is 670 million rupees towards R&D activities and plan schemes.
- Many of the departments of the organization are ISO 9001 2008 certified.
- The organization extends its expertise and facilities to other R&D sectors and industries. It also has collaboration with other R&D organizations and educational institutions. It has also identified the knowledge gap areas, where expertise needs to be developed.
- The organization, in its journey to excellence, has achieved several scientific and technological milestones, with international benchmarks and high impact on its mission programme. To quote its Director "Enhancing quality and commitment of human resources is the key to our strategy of achieving and sustaining excellence. We believe in innovations in management of science and technology for enhancing our focus, creativity and productivity" (IGCAR, 2007, p. 2) [10].

3. STUDY

The study was carried out by survey. The questionnaire contained 21 factors that can influence knowledge management maturity, which were developed based on the discussions in Bukowitz and Williams (1999) [8]. The presence of those factors can act as enablers and the absence can act as inhibitors. Also, other studies have identified similar inhibitors. For example, according to a study on knowledge management by the German Fraunhofer Institute for Industrial Engineering, the top three barriers to successful knowledge management are: lack of time, lack of awareness of knowledge management and lack of awareness of knowledge.

Bullinger et al. (1997) (as cited by Leistner (2003) [11]) and Davenport and Probst (2002) [12] classify the barriers to sharing knowledge as: personal (lack of time or confidence), collective (in-house competition), structural (poor IT infrastructure) or political (lack of openness). Chase (1997) [13] identified the key inhibiting factors for knowledge management as organizational culture, information and communication technology, organizational structure, top management commitment, non-standardized processes, emphasis on individual rather than team, incentive systems, physical layout of work spaces and staff turnover.

The questionnaire used a five point Likert scale (strongly agree, agree, neither agree/nor disagree, disagree, strongly disagree) to classify the responses from the participants. The questionnaire was pre-tested on a few senior, middle and junior level employees to gauge their understanding of the questions and the concepts that are represented. Also, since the questionnaire was personally administered by the researcher after an awareness seminar, the necessary clarifications could be provided. However, the clarifications required were minimal.

If the answer to the question is 'strongly agree', it indicates that the particular inhibiting factor is a strong one. If the answer is 'strongly agree' or 'agree' it is considered as a positively answered question (Kulkarni and Freeze, 2004) [14]. The mean, standard deviation and percentage of positive responses of typical groups for various inhibiting factors are summarized in Table 2. The strongest inhibiting factor, based on the mean for the groups, is highlighted. Detailed analysis is presented in the next section.

4. ANALYSIS

The analysis was carried out based on various groups of the organization and organization as whole to identify the prominent inhibiting factors and the corresponding enabling factors. It is obvious from Table 2 that different groups have varying intensities with regard to the different inhibiting factors. For example, for group G1, the inhibiting factor with highest intensity is 'lack of feedback', for G2 and G4 it is 'lack of integration of the process of contribution with day-to-day work' and for G3 it is 'lack of awareness of the process of contribution'. It implies that different groups need to adopt different strategies in order to improve the effectiveness of their knowledge management practices.

The analysis of one typical group is given below. The inhibiting factors of group G1, in descending order of the mean, are given in Table 3. It can be observed that for group G1, the highest intensity inhibiting factor is 'lack of feedback' with 100% positive responses, followed by 'lack of gratitude' with 78% positive responses. Also, it can be observed that the inhibiting factor 'lack of awareness of the process of contribution' (S.No.8) has a mean of 4.11 and a 100% positive response. A rearrangement of the table in the descending order of positive responses will indicate the severity of the inhibiting factor based on positive responses. The severity based on the mean indicates that more participants 'strongly agree' with respect to that inhibiting factor. The severity based on positive response indicates that more participants either 'strongly agree' or 'agree' with respect to that inhibiting factor. The respective groups need to decide whether the mean or positive response or some combination of them is the criterion for deciding the severity of the inhibiting factors.

Similar analysis was carried out for all the groups. However, owing to lack of space, the results are not given in this paper.

TABLE 2. INHIBITING FACTORS OF GROUPS

S.No.	Inhibiting factors	G1			G2			G3			G4		
		M	SD	PR	M	SD	PR	M	SD	PR	M	SD	PR
1	Lack of time	3.33	1.12	56	3.43	1.01	54	3.86	0.92	78	4.13	0.81	87
2	Lack of awareness of knowledge requirements	3.67	1.12	75	3.94	0.73	77	3.95	1.00	81	4.31	0.48	100
3	Lack of expertise in organizing the available knowledge	4.33	1.12	78	3.40	1.01	51	3.70	0.91	70	3.88	0.62	75
4	Lack of awareness of the process of contribution	4.11	0.33	100	3.66	0.84	60	4.08	0.72	84	4.00	0.52	87
5	Lack of assistance in contribution	3.22	1.39	44	3.20	1.21	43	3.43	1.12	54	3.56	0.89	56
6	Lack of user friendly technology infrastructure	3.67	1.32	67	4.06	0.76	80	4.00	0.78	75	4.31	0.70	87
7	Lack of integration of the process of contribution with day-to-day work	3.11	1.45	56	4.09	0.70	86	3.92	0.89	67	4.44	0.73	87
8	Lack of awareness of the utility of the contributions	4.22	0.67	89	3.40	1.03	54	3.57	1.01	57	3.44	1.03	62

TABLE 2. INHIBITING FACTORS OF GROUPS (cont.)

S.No.	Inhibiting factors	G1			G2			G3			G4		
9	Lack of tangible reward	3.78	1.09	56	3.00	1.08	34	3.03	1.21	32	2.94	1.06	31
10	Lack of recognition	4.22	1.20	67	3.29	0.99	48	3.14	1.13	38	3.06	1.29	50
11	Lack of gratitude	4.44	1.13	78	3.03	0.89	28	3.22	1.20	40	3.13	1.20	37
12	Lack of feedback	4.56	0.53	100	3.77	1.06	74	3.89	0.97	70	3.88	0.96	81
13	Lack of weight attached to contribution in performance appraisal	2.89	1.36	33	3.31	1.11	46	3.35	1.14	46	3.31	1.45	62
14	Lack of protection of intellectual property	4.11	1.27	78	3.40	3.80	48	3.41	1.01	48	2.75	1.34	31
15	Lack of assurance against negative reverse impact	3.89	0.78	67	3.80	0.93	71	3.84	0.90	67	3.31	1.14	50
16	Lack of assurance against belittling by colleagues	3.22	0.97	33	2.80	1.13	28	3.27	0.90	38	2.69	1.08	18
17	Lack of awareness on the significance of the contribution to the organization	4.11	1.05	78	3.77	1.03	66	3.73	1.07	62	3.88	1.15	75
18	Lack of directive from the reporting officer	3.11	0.93	33	3.06	1.00	31	3.05	0.97	38	3.50	1.15	50

TABLE 2. INHIBITING FACTORS OF GROUPS (cont.)

S.No.	Inhibiting factors	G1			G2			G3			G4		
19	Lack of contributions from colleagues	3.67	0.87	67	3.00	1.06	34	2.97	1.07	27	3.00	1.26	31
20	Lack of assurance on meeting the knowledge requirements by the organizational knowledge repository	4.00	0.76	78	3.43	1.09	46	3.54	0.96	57	3.50	0.86	75
21	Lack of mandatory organizational policy on contributions	4.00	0.7	78	3.26	1.07	74	3.14	1.25	43	3.00	1.41	31

M: mean, SD: standard deviation, PR: positive response (%)

TABLE 3. INHIBITING FACTORS OF G1

S.No.	Inhibiting factors (G1)	M	SD	PR
1	Lack of feedback	4.56	0.53	100
2	Lack of gratitude	4.44	1.13	78
3	Lack of expertise in organizing the available knowledge	4.33	1.12	78
4	Lack of recognition	4.22	1.2	67
5	Lack of awareness of the utility of the contributions	4.22	0.67	89
6	Lack of protection of intellectual property	4.11	1.27	78
7	Lack of awareness on the significance of the contribution to the organization	4.11	1.05	78
8	Lack of awareness of the process of contribution	4.11	0.33	100
9	Lack of assurance on meeting the knowledge requirements by the organizational knowledge repository	4.00	0.76	78
10	Lack of mandatory organizational policy on contributions	4.00	0.7	78
11	Lack of assurance against negative for reverse impact	3.89	0.78	67
12	Lack of tangible reward	3.78	1.09	56
13	Lack of user friendly technology infrastructure	3.67	1.32	67
14	Lack of awareness of knowledge requirements	3.67	1.12	75
15	Lack of contributions from colleagues	3.67	0.87	67
16	Lack of time	3.33	1.12	56
17	Lack of assistance in contribution	3.22	1.39	44
18	Lack of assurance against belittling by colleagues	3.22	0.97	33
19	Lack of integration of the process of contribution with day-to-day work	3.11	1.45	56

TABLE 3. INHIBITING FACTORS OF G1 (cont.)

S.No.	Inhibiting factors (G1)	M	SD	PR
20	Lack of directive from the reporting officer	3.11	0.93	33
21	Lack of weight attached to contribution in performance appraisal	2.89	1.36	33

The inhibiting factors of the organization in the descending order of the mean are depicted in Table 4.

TABLE 4. INHIBITING FACTORS OF THE ORGANIZATION

S.No.	Inhibiting factors	Mean	SD	PR
1	Lack of user friendly technology infrastructure	4.10	0.81	77.55
2	Lack of integration of the process of contribution with day-to-day work	4.08	0.85	78.36
3	Lack of awareness of knowledge requirements	4.03	0.87	82.04
4	Lack of awareness on the significance of the contribution to the organization	4.01	0.99	75.10
5	Lack of feedback	4.00	0.85	77.95
6	Lack of awareness of the process of contribution	3.89	0.80	73.77
7	Lack of time	3.72	1.00	68.16
8	Lack of assurance against negative reverse impact	3.71	1.02	64.08
9	Lack of expertise in organizing the available knowledge	3.71	0.92	64.89
10	Lack of awareness of the utility of the contributions	3.68	0.95	64.48
11	Lack of assurance on meeting the knowledge requirements by the organizational knowledge repository	3.58	0.98	56.79
12	Lack of protection of intellectual property	3.50	1.17	53.87

TABLE 4. INHIBITING FACTORS OF THE ORGANIZATION (cont.)

S.No.	Inhibiting factors	Mean	SD	PR
13	Lack of recognition	3.35	1.12	48.57
14	Lack of weight attached to contribution in performance appraisal	3.34	1.19	51.42
15	Lack of assistance in contribution	3.31	1.12	48.97
16	Lack of directive from the reporting officer	3.27	0.98	40.00
17	Lack of gratitude	3.17	1.10	37.14
18	Lack of tangible reward	3.12	1.14	36.32
19	Lack of mandatory organizational policy on contributions	3.09	1.16	37.55
20	Lack of contributions from colleagues	3.09	1.07	34.69
21	Lack of assurance against belittling by colleagues	3.00	1.05	31.02

SD: standard deviation, PR: positive response (%).

The most predominant inhibiting factors based on mean (mean $\geq$ 4.0) are 'lack of user friendly technology infrastructure' (mean = 4.10), ' lack of integration of the process of contribution with day-to-day work' (mean = 4.08), 'lack of awareness of knowledge requirements' (mean = 4.03), 'lack of awareness on the significance of the contribution to the organization' (mean = 4.01), and 'lack of feedback' (mean = 4.00).

In the positive direction, the enabling factors are given in Table 5, in descending order of vitality. It can be observed that the most predominant inhibiting factor is the most needed enabling factor.

TABLE 5. ENABLING FACTORS OF THE ORGANIZATION

S.No.	Enabling factors	Mean	SD	PR
1	User friendly technology infrastructure	4.10	0.81	77.55
2	Integration of the process of contribution with day-to-day work	4.08	0.85	78.36
3	Awareness of knowledge requirements	4.03	0.87	82.04
4	Awareness on the significance of the contribution to the organization	4.01	0.99	75.10
5	Feedback	4.00	0.85	77.95
6	Awareness of the process of contribution	3.89	0.80	73.77
7	Adequate time	3.72	1.00	68.16
8	Assurance against negative reverse impact	3.71	1.02	64.08
9	Expertise in organizing the available knowledge	3.71	0.92	64.89
10	Awareness of the utility of the contributions	3.68	0.95	64.48
11	Assurance on meeting the knowledge requirements by the organizational knowledge repository	3.58	0.98	56.79
12	Protection of intellectual property	3.50	1.17	53.87
13	Recognition	3.35	1.12	48.57
14	Weight attached to contribution in performance appraisal	3.34	1.19	51.42
15	Assistance in contribution	3.31	1.12	48.97
16	Directive from the reporting officer	3.27	0.98	40.00
17	Gratitude	3.17	1.10	37.14

TABLE 5. ENABLING FACTORS OF THE ORGANIZATION (cont.)

S.No.	Enabling factors	Mean	SD	PR
18	Tangible reward	3.12	1.14	36.32
19	Mandatory organizational policy on contributions	3.09	1.16	37.55
20	Contributions from colleagues	3.09	1.07	34.69
21	Assurance against belittling by colleagues	3.00	1.05	31.02

The following are some of the suggestions made by the participants to the open ended question, which indicates the areas that need to be improved: “some persons from each section should be made responsible to collect and make the information available”; “all organizational publications should be made available in the portal”; “provision for marks for documents and contributors should be available”; “discussion forums should be made available”; “in addition to approved knowledge documents, unapproved knowledge documents, blogs are also to be made available”; “search capability may be made more versatile”; “more credit should be given to the knowledge sharer”; “submission of knowledge documents should be made mandatory”; “record of important discussions and talks should be made available”.

5. CONCLUSION

The importance of knowledge and its management are recognized by many organizations, including nuclear organizations. However, many enabling factors are vital for the steady progress in the knowledge management journey. The acceptance of this fact by the IAEA is also evident from the following statement, “Though the importance of knowledge management in the safe and efficient operation of nuclear power plants has been increasingly recognized in recent years, the effective sharing of knowledge continues to be a challenge and many staff may be unaware of the existence of even explicit knowledge/information, let alone tacit knowledge held by other staff, which would assist them in the effective discharge of their duties” (IAEA, 2006) [15].

This study revealed the necessary enabling factors needed to attain higher levels of knowledge management maturity in a fast reactor R&D organization and its various groups. The active participation and lively discussions during the awareness seminar and survey are testimony to the willingness and eagerness of the ‘people’ involved in the knowledge management contributions and are necessary to reap the benefits. The significant differences among the various groups in the requirement of various enabling factors need further analysis. Systematic efforts to ensure the availability of these enabling factors are vital to attaining higher levels of maturity.

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A NEW IMPETUS FOR E&T ON FAST NEUTRON REACTORS IN EUROPE: INCENTIVES, STATUS, PERSPECTIVES

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Abstract

Nuclear education and training (E&T) is a key factor for safe and efficient nuclear energy production. The nuclear energy sector is facing an urgent need to replace its retiring workforce and to develop competences for the safe and secure operation of new built, long term reactor operation, decommissioning or geological disposal programmes. While today, FNRs, known as FBRs, represent less than 0.2% of the worldwide installed capacity, the GIF initiative has created a new context for innovation in nuclear systems, reopening the range of technologies to be considered in the future. The FNRs (SFR, LFR, GFR) play a central role for their potential for sustainability. The attractive and challenging scientific topics associated with the design of innovative FNRs create a new and highly incentive context for students and young scientists to embark on a nuclear career. Euratom initiatives have been taken to foster the awareness of FNR potential in a joint R&D and E&T context and stimulate the development of courses. This is done in the general framework of a normalization of nuclear E&T standards aimed at implementing optimized training schemes well fitting the required job profiles and covering the qualification of knowledge, skills and attitudes. Building on the existing sodium school in France, continuously operated over decades up to the present-day, and on the Phénix simulator, new courses on FNRs have been recently implemented. The further development of E&T infrastructures (computer codes, simulators, experimental facilities and research reactors) has been initiated in order to set up perennial E&T platforms for FNR competence qualification, especially for SFRs and LFRs. The mobility of young scientists and the mutual recognition of competences promoted in Europe are subject to widening the access to FNRs courses and skill development infrastructures in Europe and beyond.

1. THE END OF THE STORY?

1.1. From LMFBR and FBR to innovative FNRs

Today, LWRs represent more than 85% of the nuclear electricity capacity in the world. The other nuclear technologies on the market for electricity production are GCRs, PHWRs, LWGRs and FBRs.

In the past, other families of nuclear reactors have known some development because of their specific potential at improving the level of performance (HTR) and/or the utilization of natural uranium resources (MSR). The HTR and MSR, after proving some hopes in the 1960s to 1980s with the operation of demonstration reactors and prototypes, have seen their development at the industrial level postponed.

Today, FNRs (known as sodium cooled FBRs or LMFBRs) are only represented by the BN-600 operated in the Russian Federation (0.15% of the total installed capacity). In Europe, the ‘FBR phylum’, after being the subject of strong interest in several countries (France, Germany, the United Kingdom), has rapidly declined, coming to ‘extinction’ after Phénix final shutdown in 2009.

The Generation IV International Forum (GIF), launched in 2001 by the US Department of Energy, opened a renewed context for the long term development of nuclear energy. The major goals set out in the GIF roadmap [1] are in the areas of sustainability, economics, safety and reliability, and proliferation resistance and physical protection. The sustainability goals of GIF encompass both more effective fuel utilization and minimization of waste. A significant outcome of GIF was the emergence of a consistent approach, taking jointly into account both reactor and fuel cycle issues. The GIF goals were used to guide the selection of six systems for further collaborative R&D (SFR, LFR, GFR, VHTR, SCWR and MSR).

In the original GIF selection, FNRs were represented by three concepts: SFR, LFR and GFR. The R&D attempts at designing viable fast spectrum versions of the SCWR have been rather unsuccessful. Since 2005, R&D on the MSR has focused on a fast spectrum version (MSFR) presented as a promising but still long term concept. Although FNRs have been operated in the past (especially in Europe), today’s safety, operational and competitiveness standards require the design of a new generation of reactors.

1.2. E&T as a component for FNR development in Europe

Europe has defined its own strategy and priorities for FNRs: the SFR as a proven concept, as well as the LFR and the GFR as alternative technologies [2].

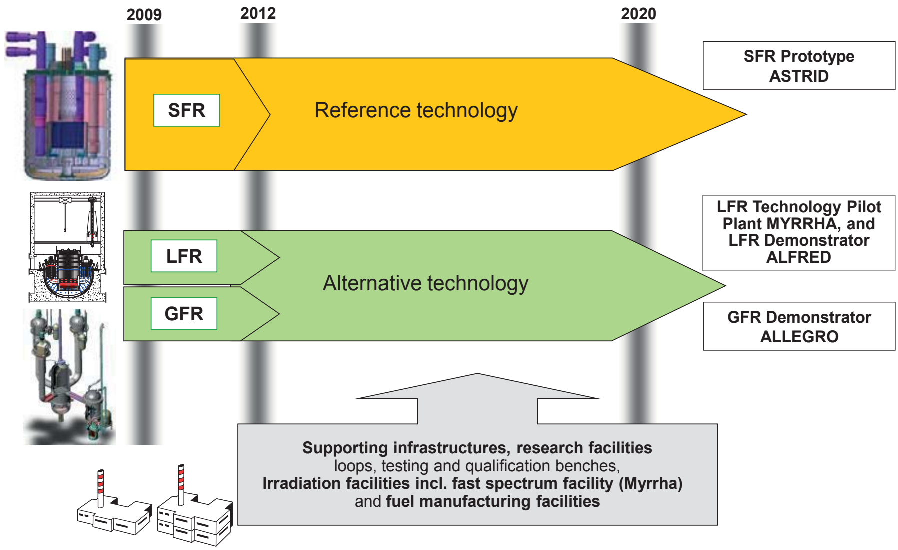

FIG. 1. Demonstration SFR, LFR and GFR projects in Europe.

There is a large uncertainty on the deployment schedule of 4th generation systems, all criteria combined. The calendar does not show an emergency in the development of fast neutron reactors in terms of uranium resources. However, the preparation of this deployment means that countries and industry will remain in the race in the coming decades. From this point of view, their involvement in R&D and in education and training (E&T) are seen as prerequisites to being a major actor at the right time.

Furthermore, the design and construction of FNR demonstration reactors or prototypes in Europe has been initiated (Fig. 1) and must be accompanied by human resource development plans supported by an efficient E&T strategy [3].

SFRs, LFRs and GFRs, seen as a renaissance of older sodium cooled FBRs in Europe, must be considered today as innovative concepts and, in that sense, generate similar requirements with respect to the viewpoint of E&T and knowledge management. A joint or coordinated effort can be foreseen.

2. FROM RECOMMENDATIONS TO IMPLEMENTATION OF E&T IN EUROPE

2.1. Nuclear E&T: Issues, concerns and trends

For a country to embark on a nuclear power programme, or continue to develop an existing programme, clear and sustained policy support from the national government is a prerequisite. Among challenges to a major expansion of nuclear capacity, developing the necessary skilled human resources is an essential issue [4].

The OECD Nuclear Energy Agency (NEA) first published in 2000 Nuclear Education and Training: Cause for Concern? [5], which highlighted significant issues on the availability of human resources for the nuclear industry. In 2012, Nuclear Education and Training: From Concern to Capability [6] considers what has changed in that time.

The following concerns should be underlined:

- Human resources could soon be facing serious challenges in coping with existing and potential new nuclear facilities. This is exacerbated by the increasing rate of retirement and a lack of attractiveness of nuclear sciences to students.
- The access and use of large research infrastructure (major experiments and state of the art computer codes) is highly relevant for E&T purposes. However, many of the experimental facilities are ageing and will soon have to be refurbished or replaced.
- A job taxonomy could be a basis for addressing the needs of workers across the nuclear sector. This is a way of enhancing mutual recognition and increasing consistency of E&T for both developed and developing countries.

These topics are briefly discussed hereafter and put into a European perspective, taking into account the EHRO-N analysis [7]. Their relevance to E&T on FNRs is the subject of Sections 3, 4, 5, and 6.

2.1.1. The nuclear job market

Although there is a lack of detailed numerical data at the national and global levels, existing surveys conducted in a number of countries suggest that future demand for global employment in nuclear related activities are in the tens to hundreds of thousands of skilled workers [6].

According to the EHRO-N survey [7], the demand for nuclear experts by the nuclear energy sector in the EU-27 is estimated, on average, to be 4000 per year up to 2020. The supply of students having been given a nuclear background (between 1800 and 2800 in the EU-27 who graduated in 2009) covers some 45–70% of the demand.

A more detailed gap analysis is needed to identify the needs for specific competences that are potentially in shortage or that need to be developed for the future. This applies especially for the areas of Gen IV and geological disposal.

2.1.2. Infrastructures for E&T

Research reactors, critical assemblies and thermohydraulic facilities can have multiple uses. Indeed, they can be used to carry out research, provide services and contribute to education and/or training, including the preparation of theses and dissertations. The status of research facilities for E&T purposes in Europe can be found in the 2012 NEA survey [6]. They are mostly devoted to LWR conditions.

Computer simulations can enhance the theoretical understanding of physical phenomena. The use of simulators in training is mandatory in some countries for the training of reactor operators.

2.1.3. Normalization of E&T standards

In Ref. [6], nuclear job specifications have been produced for the main activities associated with the construction, operation and decommissioning of commercial and research reactors, drawing up on analyses conducted by a number of companies.

A normalization of E&T standards in the nuclear sector is in progress at the European level. The initiative for a European Credit system for Vocational Education and Training (ECVET) and the stepwise introduction of a European proficiency passport are promising approaches in that respect.

According to the European Credit system for Vocational Education and Training definition, professional qualification is expressed in terms of knowledge, skills and competences that can be assessed, validated and, possibly, certified [8–10]. Learning outcomes means statements of what a learner knows, understands and is able to do on completion of a learning process and which are defined in terms of knowledge, skills and competences.

2.2. The pedagogical dimension, from job profiles to training schemes

In a context of normalization of E&T standards in Europe, a number of Euratom Fission Training Schemes have been launched aimed at structuring career development across the EU. The ultimate objective of each Euratom Fission Training Scheme is to develop a European competence passport. The 7th Framework Programme (FP7) of the EU targets research into sustainable energy and security of supply. Five Euratom Fission Training Schemes lie in this area [6].

ENEN-III are training schemes to upgrade knowledge and develop skills as required by specific positions for nuclear system suppliers. These comprise four levels: basic nuclear topics for non-nuclear engineers, design challenges for Gen III nuclear power plants, construction challenges for Gen III nuclear power plants, and design challenges for Gen IV nuclear power plants.

ENETRAP-II is aimed at developing European high quality 'reference standards' and good practices for E&T in radiation protection, specifically with respect to the radiation protection expert and the radiation protection officer.

PETRUS-II, Programme for Education Training Research on Underground Storage, focuses on the competences required by radioactive waste agencies for professionals working on geological disposal. In this scheme, a Science and Technology Passport is being developed.

TRASNUSAFE is aimed at devising two training schemes on nuclear safety culture within a European environment. On the basis of the evaluation of the specific training needs across Europe, the training schemes will include a common generic basis module and four specialized modules that will be validated by means of pilot sessions.

CINCH, Cooperation in Education in Nuclear Chemistry, provides a virtual learning platform for collaborative modular postgraduate development.

Of particular interest for the FNR context is the ENEN-III project where one of the addressed job profiles is the development and pre-conceptual design of Gen IV nuclear reactors [11]. A brief summary is given in Section 3.4.

3. E&T AND FAST NEUTRON REACTORS: WHAT ARE THE ISSUES AND THE NEEDS?

3.1. A specific FNR context? Concerns and challenges

Recent international surveys [6, 7, 12] have underlined the alarming character of the situation. However, the specific aspects of E&T in support of FNR development do not seem to have been explicitly addressed.

The concerns and challenges of E&T to prepare, ad hoc, a competent workforce to support nuclear industry are basically the same for FNRs as the currently operated and constructed (or planned) LWRs (cf. Section 2.1):

- A shortage in human resources of qualified personnel, exacerbated by the increasing rate of retirement and a lack of attractiveness of nuclear sciences to students;
- The availability and access to research infrastructures for E&T purposes;
- A normalization of E&T standards, emphasizing the development of skills and competences, in addition to knowledge, and tightly accounting for the needs of the nuclear job market.

A human resource plan still has to be produced, taking into account the feedback experience gained in the design and operation of past FNRs, together with LWRs.

The general trends are:

- The decommissioning programmes expected over the next two/three decades;
- The preparation of demonstration and development of innovative reactor technologies, including FNRs of 4th generation.

However, some specific concerns are attached to E&T for FNRs. Focusing on the European context:

- The timeframe extending over several decades and covering a very significant design phase (innovative FNR options), in preparation for the construction and operation phases;
- The small number of nuclear reactors to be built in the mid-term (one to three demos or prototypes within 2020–2030);
- Qualitative aspects such as the high degree of innovation (many design options remain open today), the compliance to renewed E&T standards, the enhanced recourse to practicals in training (best estimate codes, simulators, experimental facilities and reactors).

The present and near future job market of Gen IV nuclear engineers is strongly based, on the one hand, on the decisions concerning large demonstration facilities and on the other hand on the replacement and supplementing of personnel in research facilities active in the field. Presently, a detailed, reliable estimate of the numbers of experts and their profiles cannot be made. The main need is probably for research oriented people, either with profound nuclear engineering experience or, on the other hand, high level experts in specific areas complemented by a sufficient basic knowledge in nuclear engineering.

3.2. The attractiveness of FNRs to students and scientists

For most countries engaged or engaging a nuclear production programme, fast reactors are not the priority type of facilities to build. However, convincing young students with high potential to embark on a nuclear career requires attractive and challenging scientific topics.

The Gen IV nuclear reactors are characterized by higher operating temperatures. High temperature materials, corrosion effects, liquid metal dynamics and heat exchangers are typical topics. The Gen IV nuclear reactors are also characterized by fast neutron fluxes for both breeding and enhanced burning of long lived waste products. Another topic is the development and testing of entirely new nuclear fuels and fuel cycles, together with new fuel fabrication and fuel recycling concepts.

The Gen IV studies offer such topics and open up new prospects for sustainable nuclear energy that this new generation of scientists may see during their lifetime. Having already benefited from several prototypes experimentations, fast reactors are offering a greater scope in actual issues, such as improved design features, fuel options and performance, deployment strategy and insertion in a LWR park, ability to minimize nuclear waste, etc. These topics may take the form of group works, short term internships or PhD studies in R&D laboratories.

Beyond this educational merit, young engineers investing for some years in fast reactor studies may take benefit from it when working later on LWRs, through a broader expertise and approach on some transverse areas such as safety, core physics, engineering, materials, etc.

3.3. What is the job profile for designing Gen IV nuclear reactors?

The (pre-)conceptual design of a new concept of nuclear reactor is a ‘systemic’ and interdisciplinary task requiring several types of qualification profile.

A design team would be composed of specialists in the various scientific disciplines (such as neutronics, thermo-hydraulics, material science, coolant technology, etc.) and ‘assembling’ engineers capable of performing the optimized integration of topical results into realistic reactor components and balance of plant (Fig. 2).

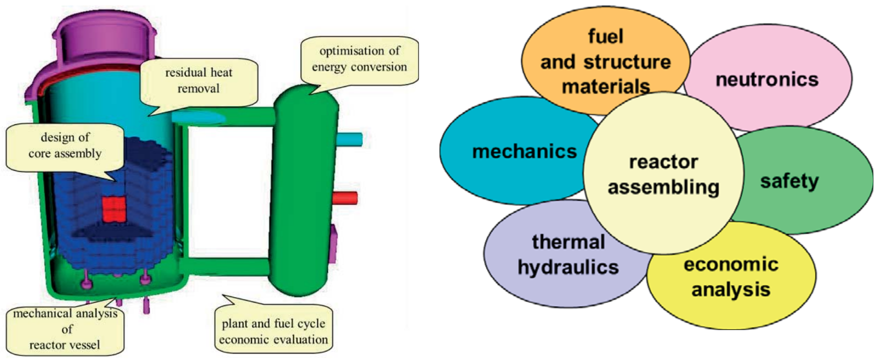

FIG. 2. Systemic and interdisciplinary approach for innovative design.

As far as the 'assembling' task is concerned, a comprehensive account of the multiple interactions between design parameters is not feasible, even for the best experts. Optimizing the design of a nuclear reactor core is a very complex task. Currently this is done by successive iterations between different disciplines.

The overall parametric and interdisciplinary optimization of a reactor concept is the subject of development. See, for example, the FARM (FAst Reactor Methodology) tool developed by the CEA [13] for optimizing the core performance and safety characteristics of GFR cores. This approach could be used as an E&T tool to develop skills in designing innovative reactors.

3.4. Implementing training schemes for the design of Gen IV nuclear reactors

In the FP7 ENEN-III project, an attempt has been made to define a training scheme for the development and pre-conceptual design of Gen IV nuclear reactors (Training Scheme D, or TSD) [11].

All six Gen IV reactor types have been considered in this training scheme (SFR, LFR, GFR, VHTR, SCWR and MSR). However, more emphasis has been put on the three reference FNR concepts selected in the Euratom strategy (SFR, LFR, GFR).

A training scheme for the design of Gen IV nuclear reactors will be more research oriented and will have a broader and less specialized scope than training schemes on Gen II or Gen III reactors. It is expected to respond to the current needs of the research communities in order to design (short to mid-term), build and operate (mid- to long term) the demos and prototypes of the nuclear reactors of the future (ASTRID, ALFRED, ALLEGRO).

Some of the basic principles and introductory courses are common to all Gen IV concepts but, in deeper expertise levels, the specific reactor type or the activity areas of the support facilities have to be selected.

The classification of learning outcomes according to domains of knowledge and areas of interest is shown in Table 1.

TABLE 1. STRUCTURE OF ENEN-III TSD (DOMAIN OF KNOWLEDGE (K))

Domains of knowledge (K)	TSD	Areas of interest
General knowledge on Gen IV systems and technology	TSD.K001	Introduction to Gen IV systems and technology
	TSD.K002-1	Introduction to the SFR
	TSD.K002-2	Introduction to the LFR
	TSD.K002-3	Introduction to the GFR
	TSD.K002-4	Introduction to the VHTR
	TSD.K002-5	Introduction to the SCWR
	TSD.K002-6	Introduction to the MSR
	TSD.K003	General safety features of Gen IV systems
	TSD.K004	Structural materials for Gen IV reactors
	TSD.K005	Fuels for Gen IV reactors
	TSD.K006	Gen IV and the closed fuel cycle
Design specific knowledge for the SFR	TSD.K101	SFR core design
	TSD.K102	Structural materials challenges for SFR
	TSD.K103	SFR (primary circuit) design
	TSD.K104	Instrumentation techniques for SFR
	TSD.K105	Safety issues related to using sodium as a coolant
Design specific knowledge for the LFR	TSD.K201	LFR core design
	TSD.K202	Structural materials challenges for LFR
	TSD.K203	LFR primary circuit design
	TSD.K204	Instrumentation techniques for LFR
	TSD.K205	Safety issues related to using lead as a coolant
Design specific knowledge for the GFR	TSD.K301	GFR core design
	TSD.K302	Structural materials challenges for GFR
	TSD.K303	GFR primary circuit design
	TSD.K304	Instrumentation techniques for GFR
	TSD.K305	Safety issues related to using helium as a coolant

According to experts, engineers involved in research into Gen IV reactor types need to have a basic training on the general aspects of Gen IV systems and technology. This knowledge area, relatively independent of the specific reactor type, also covers critical cross-cutting areas such as safety, structural materials, fuels and fuel cycle.

Next to this, for each reactor type, the learning outcomes for design specific challenges were elaborated separately. Design specific knowledge has been treated similarly for each Gen IV reactor type, focusing on the following five areas of interest: core design, material challenges, primary circuit design, instrumentation techniques and safety issues related to the coolant.

Learning outcomes, not shown in the present paper, are documented in Ref. [11].

4. THE SODIUM SCHOOL AND THE PHENIX SIMULATOR

Since its inception, Phénix has been a joint programme between the French Atomic Energy Commission (CEA) (80%) and Electricité de France (EDF) (20%). Both partners contributed proportionally to the plant's operating budget. The personnel (approximately 280 persons) were composed of mixed teams [14].

The French Sodium School [15] and the SIMFONIX Phénix simulator [14] have been key E&T instruments for the training of personnel involved in the operation, safety and dismantling of historical FBRs (Phénix, Superphénix).

4.1. The French Sodium School

The French Sodium School was created in 1975 at the Cadarache Research Centre for the training of Phénix plant teams. It was also used to train sodium loop technicians in support of R&D activities. It was accredited by EDF in 1980 for the training of Superphénix plant teams. In 1998, with the decision to shut down Superphénix, the sodium school became more oriented towards decommissioning activities [15].

Trainees usually belong to French nuclear stakeholders (CEA, EDF, AREVA, IRSN), and other companies involved in sodium activities (belonging or not to the nuclear industry).

Since its creation, the French Sodium School has been open to foreign countries. Specific training sessions have been organized for different FBR operators (Germany, Japan, UK) and in support of decommissioning actions (DFR and PFR in the UK). More recently, the school, in partnership with the Phénix plant, has widened its audience to China (CIAE), India (IGCAR, for PFBR safety issues) and the Russian Federation (Rosatom).

The French Sodium School addresses all aspects of the safe operation of sodium facilities and their decommissioning, with different modules dealing with:

- Practice of sodium purification;
- Practice of sodium circuit operation;
- Safety and management of sodium risk;
- Practice of intervention on sodium circuits;
- NaK management and safety;
- Dismantling of sodium installations.

The sodium school makes extensive use of tutorials (case studies) and practicals (dedicated cells and facilities, instrumented devices).

Since 1975, more than 4500 people have been trained at the French Sodium School. The present activity corresponds to about ten sessions per year for an average of 130 trainees.

4.2. Fast reactor operation and safety simulator (SIMFONIX)

SIMFONIX is a system that simulates the basic principles of the Phénix power plant. Even though it is not a full scale simulator, it allows a good display of the main parameters and interactions between physical phenomena. Consequently, it has been used for the training of Phénix personnel for reactor operation under normal and incidental conditions [14, 15].

The simulator practicals using SIMFONIX cover a large range of SFR operating conditions:

- SFR reactor kinetics and control (subcritical approach and criticality, reactivity step, feedback effects, rod calibration, reactor protection);
- SFR operation (startup, full power and load reduction operation, reactor shutdown);
- SFR incidental transients (neutronic incidents, primary cooling system incidents, secondary cooling system incidents).

The SIMFONIX simulator has been used as an FBR E&T tool for about 20 years. In 2005, SIMFONIX was integrated in the Fast Reactor Operation and Safety School (FROSS) created at the Phénix plant. To anticipate the obsolescence of the SIMFONIX tool, the development of a computer based simulator for application to the ASTRID prototype has been recently undertaken (Section 6.2).

5. E&T AND FNR: FROM KNOWLEDGE PRESERVATION TO SKILL DEVELOPMENT

5.1. The European context and the E&T actors

Euratom research and training programmes promote knowledge generation (research) and competences development (training) in a strategy combining research, innovation and education [8]. Specific E&T programmes aim at developing training schemes for the different job profiles needed in the nuclear world (Section 2.2). Euratom FP7 research projects systematically include E&T actions (Section 5.3).

The ENEN (European Nuclear Education Network Association) network (http://www.enen-assoc.org) was established in 2003 to preserve and further develop expertise in the nuclear fields through higher E&T. The ENEN currently has over 60 members, mainly in Europe. This objective is realized through cooperation between universities, research organizations, regulatory bodies, industry and many other organizations involved in the application of nuclear science and radiation protection. ENEN fosters student mobility within Europe and beyond.

France has an important nuclear teaching platform organized around engineering schools, universities, research laboratories, technical schools and also nuclear companies or entities dedicated to providing professional training. In this context, I2EN, the International Institute for Nuclear Energy (http://www.i2en.fr) [16], set up in 2010, is federating French entities and delivering high level curricula in nuclear engineering and science and is promoting the French offer for E&T in partner countries.

The INSTN, the Institut National des Sciences et Technologies Nucléaires (http://www-instn.cea.fr), with its own nuclear engineering Masters level (or specialization) degree and a catalogue of more than 200 vocational training courses, is the major nuclear E&T operator in Europe.

There is a rather extensive number of national and international courses (educational and vocational), workshops and seminars being organized in Europe dealing with Gen IV related topics, including FNRs. Without claiming a comprehensive review, an attempt has been made to categorize them according to three complementary and interlinked objectives:

- Refreshing the attractiveness of the nuclear sector by promoting the awareness of the potential of Gen IV systems, and particularly FNRs (Section 5.2);
- Stimulating the propagation of scientific knowledge on specific FNR technologies (Section 5.3);
- Consolidating efficient training schemes on FNRs for competence development (Sections 5.4 and 6).

5.2. Promoting the awareness of the potential of Gen IV systems

5.2.1. Universities and engineering schools

Universities and engineering schools should play an essential role in promoting the awareness of advanced nuclear systems.

I2EN has recorded the main curricula in France for engineering and Masters degrees [16]. A quick review of the 22 listed curricula suggests (using 'reactor design' as a topical area search term) that 50% of them include courses on the different families of nuclear reactors (present and future), therefore including FNRs (Fig. 3). A similar situation can be expected in other European countries operating educational programmes in nuclear sciences and engineering.

The time allocated to FNRs is rather limited, in the range of 1 to ~6–8 hours. This is not sufficient for students to acquire professional skills but an effective way to develop their awareness of the potential of FNRs.

5.2.2. ENEN/INSTN international course on Gen IV systems

The ENEN international course, Generation IV: Nuclear Reactor Systems for the Future (Fig. 4), should be considered as an introductory course to the issues, benefits and challenges of the six nuclear systems selected in GIF.

In 2012, the course was reconfigured in order to better match the R&D priorities of the European strategy, therefore giving a greater emphasis to innovative FNRs (SFR, LFR, GFR).

This course is expected to properly cover most of the learning outcomes relating to general knowledge of the main features of Gen IV systems and providing trainees with a critical attitude [8]. Similar courses are run or under development in other European countries.

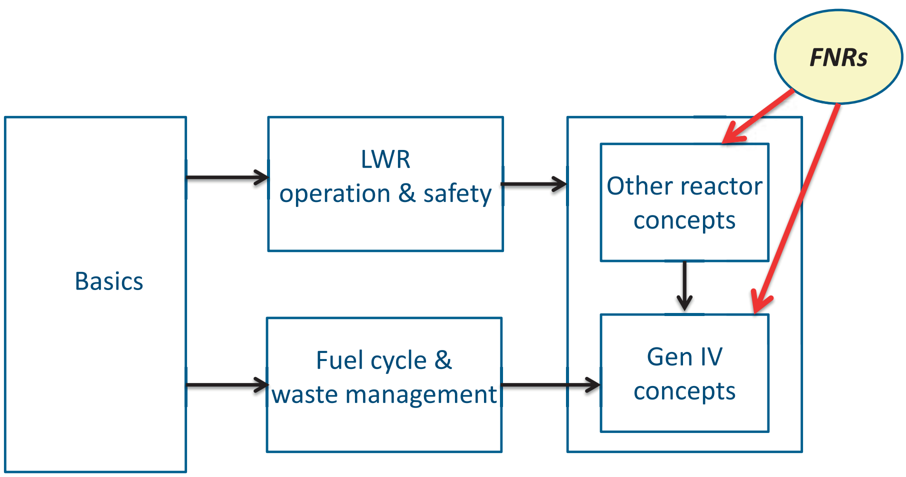

FIG. 3. Typical training scheme in nuclear engineering.

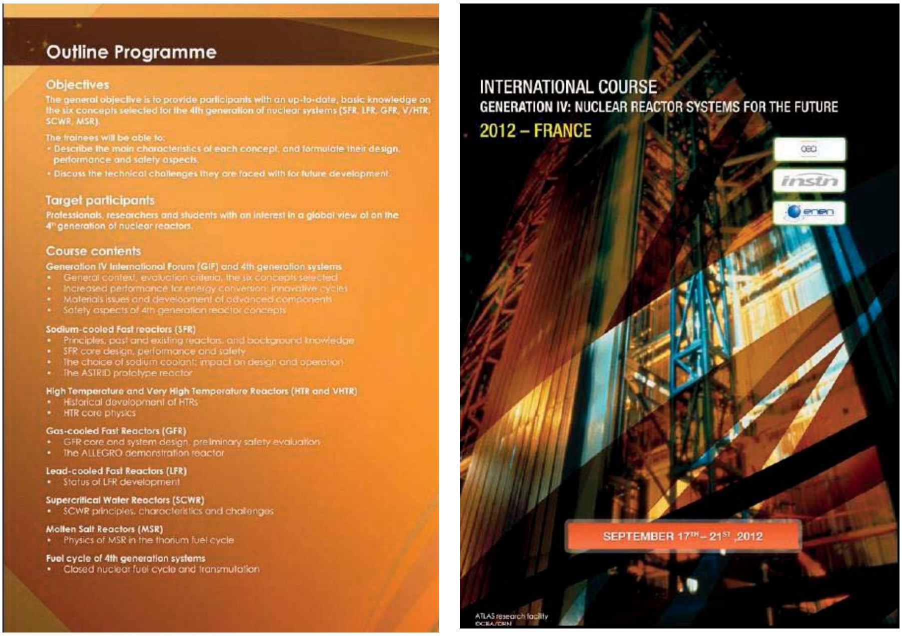

FIG. 4. Leaflet of the ENEN international course. Generation IV: Nuclear Reactor Systems for the Future (2012 edition).

This general overview of Gen IV systems can also be considered as a first step into a more complete training scheme on a specified reactor concept (Section 5.4).

5.3. Propagating scientific knowledge on specific FNR technologies

In the Euratom FP7 programme in the area of advanced nuclear systems (2007–2013) [8], R&D projects incorporate E&T tasks in the form of workshops and training courses. Workshops are focused on R&D advances while courses may have a more general content.

In general, the projects include a dedicated E&T work package or subproject which is aimed at coordinating the training of Masters and PhD students and the staging of formal courses on technologies. Such work packages enable students from the universities within the consortium to receive hands-on training through placements (internships) with other partners which are industrial and research laboratories. The workshops and training courses are open to non-partner institutions (in particular those from third countries) and generally coordinated with the ENEN network to strengthen the visibility and the impact of the actions. Workshops are focused on R&D advances while courses may have a more general content.

Some projects are concept oriented (CP-ESFR on SFR, LEADER on LFR, GOFASTR on GFR, EVOL on MSR). Other projects address cross-cutting FNR R&D areas such as materials (GETMAT on materials for advanced reactors and transmutation technologies) and nuclear fuels (PELGRIMM on transmutation fuels), while others are focused on the specific safety aspects of Gen IV systems (JASMIN for SFR, SARGEN-IV for SFR, LFR and GFR).

In the CP-ESFR project, a specific subproject (SP5) has been dedicated to E&T actions (~5% of the total budget). Five scientific workshops and one course (Functional Analysis and Design Safety of SFRs) have been held during the 4 years duration of the CP-ESFR project (2009–2012).

A similar approach has been set up in the frame of the LEADER project with WP7 dedicated to E&T actions. Students on PhD and Masters level have been trained in the science and technology pertaining to LFRs. Four workshops have been planned (the third took place in September 2012), to which students present and discuss the results of their work, with the possibility of gaining feedback from the work package leaders. The duration of each workshop is typically 3–4 days, 1–2 days devoted to lectures, 1 day to student work and 1 day to student presentations.

Another example is the recently started PELGRIMM project dedicated to the study of transmutation fuels (minor actinide-bearing fuels). Work package 6 is devoted to knowledge transfer, communication and E&T. In PELGRIMM, ENEN, directly involved as a full partner of the project, is in charge of E&T actions. The project will initiate and fund 8 internships (2 trainees per year) of Masters degrees in Europe (periods of 6 months) in order to provide incentives to young people to undertake PhD studies and join later the European R&D nuclear community. The internship subjects are proposed by the PELGRIMM partners. The ENEN network is used to identify students supported by the project for their internship. In addition, two training courses on ‘closed fuel cycle’ will be arranged by ENEN.

5.4. Implementing training schemes on FNRs

The workshops organized in the scope of Euratom FP7 projects are generally focused on R&D advances and may lack continuity. Courses are intended to become perennial E&T modules. After the initiation of the process by Euratom, it appears therefore suitable that nuclear E&T institutions (e.g. INSTN) do the work of converting them into stable courses and training schemes (pedagogical support, database of lecturers, management of course materials, QA process, including design documents and evaluation procedure, communication and logistical organization).

5.4.1. The INSTN training scheme on SFRs

The growing interest of the international nuclear community in the design of innovative SFRs, and the decision in France to build a Gen IV SFR prototype reactor by 2020 (ASTRID), have fostered a rapid increase of R&D programmes on SFR design. Building on the existing E&T framework on the FBR in France (cf. Section 4), four SFR training modules (one-week sessions) have been successively implemented between 2007 and 2010 within the frame of INSTN.

Together with the general introductory course on Gen IV systems (cf. Section 5.2), these modules form a consistent and extensive training scheme on SFRs (Fig. 5).

These modules are mainly targeted at the experts involved in the design and development of innovative SFRs, related experimental facilities and the ASTRID prototype. Table 2 and Figures 6(a)–(b) show some statistics on the registration to the different modules.

The module dedicated to the safety and operation of fast reactors combines lectures with practical sessions on the Phénix simulator (SIMFONIX, cf. Section 4.2). The module on SFR core physics optionally includes applications of the ERANOS computer code (FNR core physics).

The SFR general module is attended by the majority of trainees, at least as a first step. Participants to more specialized modules are rather equally distributed, according to background and job profile.

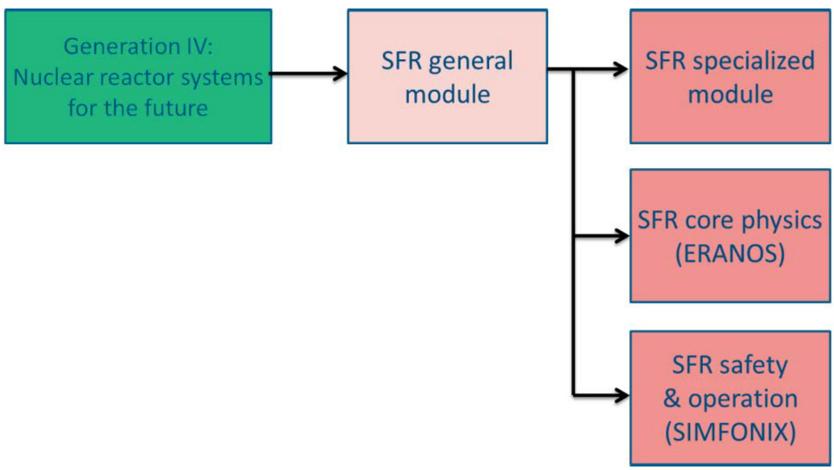

FIG. 5. Structure of the INSTN SFR training scheme.

TABLE 2. REGISTRATION FOR DIFFERENT MODULES

INSTN module	Total 2007-2012
Generation IV	38
SFR general module (*RNR-Na module général*)	157
SFR specialized module (*RNR-Na module spécialisé*)	72
SFR core physics + ERANOS (*Physique du cœur de RNR, ERANOS*)	56
SFR safety and operation (*RNR-Na fonctionnement et sûreté en exploitation*)	58
Total	381

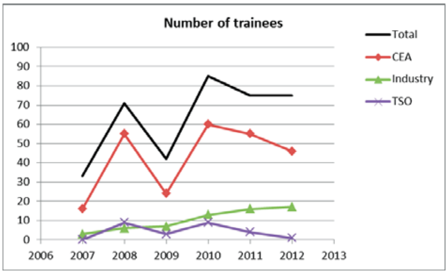

FIG. 6(a) Number of trainees in the Gen IV and SFR INSTN courses.

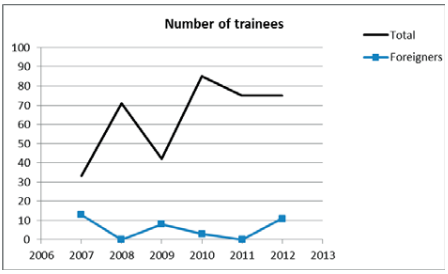

FIG. 6(b) Number of trainees in the Gen IV and SFR INSTN courses (foreigners).

The total number of trainees is self-adjusting and has remained rather stable versus time, with an average of about 60 trainees/year, the majority being CEA employees (ASTRID project team). The participation of industrial partners is increasing. The registration of foreigners is very limited, about 5 trainees/year, with a majority attending the introductory course on Gen IV systems. This can be explained by the fact that, although a specific session on SFR reactor physics was held in English in 2009, the SFR courses are presently given in French. They should be converted into English in the near future.

5.4.2. International School in Nuclear Engineering

To attract top quality national and international PhD students and researchers, the INSTN, in 2007, established the International School in Nuclear Engineering. The 2012 session consisted of 6 one-week independent PhD level courses taught in English (Fig. 7). The school is designed for PhD students but is also open to nuclear engineering researchers (~100 participants attended in the 2012 session).

These courses are not focused on FNRs but provide trainees with high level basic knowledge on the main disciplines involved in the design of nuclear reactors. LWRs are considered as the reference but the specific challenges and recent R&D developments for Gen IV reactors are elaborated. They are therefore well suited to train future experts on the specific topics involved in Gen IV reactor design.


OUTLINE PROGRAMME OF COURSES

Reactor Core Physics: Deterministic and Monte Carlo Methods
(C. Diop, A. Santamarina)
- The neutron transport equation
- The neutron diffusion equation
- Solving the transport equation
- Solution algorithms for discrete ordinates methods
- The Monte Carlo method for solving the transport equation
- Monte Carlo techniques: fixed source, variance reduction, criticality, perturbation calculations, adjoint calculation, applications to shielding

Thermal Hydraulics and Safety
(D. Bestion, J-M. Seiler, E. Studer)
- Basic modelling of two-phase flow
- Two-phase flow phenomena in LWRs
- Multi-scale approach of LWR thermal hydraulics
- System code modelling of reactor thermal hydraulics, including advanced modelling
- Simulation of LWR design basis accidents
- Application of two-phase CFD to reactor thermal hydraulic issues
- Multiphase phenomena and modelling of severe accidents in LWRs
- Hydrogen risk (production, dispersion, combustion, mitigation)

Materials for Nuclear Reactors, Fuels and Structures
(J-L. Béchade, F. Garrido, J-C. Brachet)
- Mechanisms of irradiation damage: neutrons, photons, electrons
- Behaviour of materials under irradiation: ferritic steels for reactor pressure vessel, austenitic stainless steels for internals or fuel cladding (FBR), Zr alloys for fuel cladding and fuel assemblies (LWR)
- Fuel materials (UO_2, PuO_2): microstructure evolution, thermomechanical behaviour
- Materials for high temperature conditions: SiC, ZrC, low swelling alloys
- Materials for fusion: low activation materials, resistance to high-energy neutrons, breading blankets

Nuclear Fuels for Light Water Reactors and Fast Reactors
(D. Parrat, J. Noirot)
- Nuclear fuels fundamentals
- Fuel element thermal performance and temperature effects
- Nuclear fuel behaviour under irradiation

INTERNATIONAL SCHOOL IN NUCLEAR ENGINEERING

- Main limiting phenomena in the different types of fuels
- Fuel behaviour during some off-normal conditions
- Modelling of fuel behaviour
- Fuel challenges for the future

Nuclear Fuel Cycle and Reprocessing
(B. Boullis, Ch. Poinssot, P. Baron)
- Fuel cycle and spent fuel
- Fundamentals of fuel cycle: chemistry of actinides and fission products in solution
- PUREX spent fuel processing: fundamental knowledge and industrial process
- Minor actinide reprocessing
- Advanced fuel cycles

Nuclear Waste Management
(B. Bonin)
- General considerations on nuclear waste
- Waste conditioning
- Waste storage and disposal
- Perspectives

FIG. 7. Programme of the international course in nuclear engineering (2012 session).

6. PERSPECTIVES, NEW INSTRUMENTS FOR SKILL QUALIFICATION

6.1. From technological schools to training platforms

The relevance of research infrastructures for E&T purposes (quality and attractiveness of high level training in nuclear technology) has been underlined in previous sections (Sections 2.1, 3.1, 3.2). This covers research reactors, critical assemblies, thermohydraulic facilities, fuel cycle related laboratories and state of the art computer codes, computer based simulators, etc. It is generally considered that such infrastructures should be used more systematically within nuclear education to provide students with a more direct and more personal experience of nuclear phenomena and their characteristics.

A distinction should be made, a priori, between research oriented infrastructures and more specifically E&T dedicated infrastructures. The former can be involved in the subjects of internships and doctoral theses. The latter can be used for more direct and practical E&T applications but are in very limited number in Europe and restricted to LWR conditions. In France, the ISIS training reactor of INSTN (Saclay) is rather intensively used for practicals in E&T sessions [17]. ULYSSE (Saclay) is under decommissioning. Some reactors, such as MINERVE (Cadarache), are more or less dual purpose tools for research and E&T.

In Europe today, there is no research reactor (or critical assembly) which is still operated in fast neutrons conditions. MASURCA (Cadarache) is under refurbishment and could be used in the frame of doctoral theses.

A small number of thermohydraulic facilities in Europe are dedicated to studies on flow and heat transfer phenomena involving liquid metals (sodium, lead, lead–bismuth eutectics, etc.). Participation of students is limited to doctoral theses, but one should not underestimate the importance of these facilities for the future, in particular to train technicians in the practical aspects of liquid metal utilization [6]. For new facilities, their potential for E&T purposes should be analysed at the design stage.

The FNR demonstration reactors and prototypes envisioned in Europe in the 2020–2030s (ASTRID, ALFRED, ALLEGRO) will not likely be adequate tools for E&T sessions. The development of reactors specifically designed for E&T as instruments for skill development is under discussion. An alternative is the use of well-designed computer based simulators such as SIMFONIX (cf. Section 4.2). Some initiatives under consideration are presented in Sections 6.2 and 6.3, in the frame of the ESNII+ project submitted to FP7 for evaluation by the end of 2012.

6.2. An SFR and sodium technology international school

The French Sodium School and the SIMFONIX simulator have played a major role in knowledge preservation across decades. These precursory and exemplary tools are being upgraded to take into account the new E&T needs generated by the emergence of a new generation of SFRs, and particularly ASTRID.

Owing to increasing E&T needs for the ASTRID project (from pre-conceptual design to operation), while keeping the ad hoc training capacity for FNRs in an operational phase (outside Europe) or being decommissioned, there is a need to upgrade the E&T offer for SFRs [10]. Together with the recently set up SFR training courses, the sodium school and Phénix simulator can be the basis for an international SFR platform, thus implementing a comprehensive training scheme.

6.2.1. Upgrading the French Sodium School

The training capabilities of the French Sodium School have to be enhanced. This includes the upgrading of pedagogical strategy and educational tools and the updating of the offer of training sessions. New benches, dedicated to a liquid metals physical properties demonstration laboratory, Na–H_2O interaction, sodium fire management, ultrasonic techniques and magneto-hydrodynamic effects are envisioned. These tools, developed for E&T, can also play a positive role in communication to improve public acceptance on sodium technologies.

6.2.2. Computer based simulator: From Phénix to ASTRID

Originally developed for the training of Phénix operators, the SIMFONIX simulator has recently been inserted in the new INSTN SFR offer (SFR operation and safety module). The anticipated obsolescence of the tool and the new SFR context have been strong incentives for the specification of a new SFR simulator. A particular objective is to fit the ASTRID design with different options, including the energy conversion system. The general E&T objectives remain basically the same as those of SIMFONIX (practicals in operation and safety), but also include human factor studies and human–machine interface design. The definition of the main functional specifications of the simulator is under way.

6.3. ELECTRA, a platform for research and E&T on LFRs

In order to gain experience on lead cooled reactor operation, and to provide a unique facility for E&T, the KTH is developing the ELECTRA concept (European Lead Cooled Training Reactor) [18]. The main component of ELECTRA is a 500 kW fast neutron reactor with (Pu,Zr)N fuel, cooled by natural convection of liquid lead. Associated fuel cycle facilities are also planned. ELECTRA is intended to function as a training facility in support of European fast reactors projects, such as MYRRHA and ALFRED. It will also permit carrying out highly innovative research on fast reactor dynamics and fuel cycle processes.

7. CONCLUSIONS

The nuclear community is facing a general problem of a decrease in qualified personnel due to the retirement of ageing workers and a lack of replacement workers. This is worsened by the fact that students are not enough aware of, or no longer interested in, professional opportunities in the nuclear sector.

The attractive and challenging scientific topics associated to innovative FNRs create a new and highly incentivized context for students and young scientists with high potential to embark on a nuclear career. Thus, the perspective of the construction of demonstration reactors or prototypes of SFRs, LFRs and GFRs may appear a strong driver.

Stimulated by Euratom initiatives, the E&T system in Europe is being adjusted to account for the new needs and challenges generated by the perspective of long term development of innovative concepts of FNRs. The FP7 R&D projects systematically include E&T activities (workshops and courses on specific FNR topics, funding of internships and PhDs). In parallel, E&T projects play a major role in developing a job taxonomy as a starting point to the development of standardized training schemes, for example, ENEN-III for the design of a Gen IV nuclear reactors job profile. ENEN, INSTN and I2EN are increasingly involved in the practical implementation of such training schemes.

For SFRs, an exemplary and precursory approach in France has permitted preserving the knowledge and know-how gained during five decades of R&D and for this to be passed down to future generations. The continuous operation of the French Sodium School and of the Phénix plant simulator have created a favourable context to restart E&T courses and provide tools on SFRs in response to the human resources needed to develop a new generation of SFR.

International E&T surveys have strongly underlined the complementary role of skills and competences, in addition to knowledge, for the qualification of nuclear workers. For this, E&T infrastructures (simulators, computer codes, experimental infrastructures and research reactors) are called on to play a major role to complement these courses. Building up on the existing sodium school in France, the development of new infrastructures is being considered in Europe.

The mobility of young scientists and the mutual recognition of competences promoted in Europe are subject to widening access to FNRs courses and E&T infrastructures. This is especially valuable in a context where new FNR prototypes are not affordable for most countries and where the long term objectives give time for some decades of cooperation without industrial competition.

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INFORMATION SHARING FRAMEWORK AMONG EXPERTS FOR FACILITATING DEVELOPMENT OF FAST REACTORS AND FUEL CYCLES

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Abstract

Transparency in the peaceful use of nuclear energy is important as a measure to complement and reinforce IAEA safeguards and promote international/regional confidence building. Moreover, information sharing, a key component of confidence building, will help to promote the development of fast reactors and associated fuel cycles by enhancing transparency, capacity building and encouraging understanding among non-proliferation experts. Currently, the Japan Atomic Energy Agency is carrying out a project to design and establish an information sharing framework (ISF) for supporting and promoting nuclear transparency in the Asia Pacific region. This is a cooperative effort with Sandia National Laboratories, the Korean Institute for Non-proliferation and Control and the Korea Atomic Energy Research Institute. At present, requirements for planning and implementing the ISF are under discussion to address inherent challenges that are recognized among project partners. This paper describes the current status of the requirements development process for the ISF. Though the requirements are still under development, they will be finalized and demonstrated in the near future by the project partners.

1. INTRODUCTION

Transparency, in the context of the peaceful use of nuclear energy, was defined by the Cooperative Monitoring Center at Sandia National Laboratories (SNL) as “a cooperative process of providing information to all interested parties so that they can independently assess the safety, security, and legitimate management of nuclear materials [1]”. This paper asserts that transparency is a voluntary activity, which is supplemental to obligations required by international or national agreements, to provide additional assurances [2].

The Asia Pacific region has a broad spectrum of nuclear development under way and planned in the future. In the context of the development of fast reactors and associated fuel cycle facilities, the status is also varied; several countries, such as China, India and Japan, already have fast reactors. The Republic of Korea is pursuing its plan to develop them and has completed the conceptual design of a prototype fast reactor. Meanwhile, other countries currently do not have concrete plans but might be interested in understanding more about fast reactor technologies for safety, non-proliferation and nuclear security reasons. Therefore, nuclear transparency is essential for providing additional assurance and enhancing confidence building in this area.

Currently, a joint project entitled, an Information Sharing Framework for Regional Non-proliferation Cooperation for Enhancing Nuclear Transparency (ISF) is being carried out by SNL, the Korean Institute for Non-proliferation and Control (KINAC), the Korea Atomic Energy Research Institute (KAERI) and the Japan Atomic Energy Agency (JAEA). This project, which began in 2011, is planned and implemented under bilateral R&D arrangements between the US Department of Energy (DOE) and the JAEA, and the DOE and the Republic of Koreas's Ministry of Education, Science and Technology. Though the official State-to-State project agreements are bilateral, the activities can be carried out multilaterally between the Japan, the Republic of Korea and the United States of America. The goal of the project is to design and establish a framework that enables the direct, transparent sharing of non-proliferation and safeguards relevant information among non-proliferation experts. During the planned two year effort, the objectives are to clarify the needs of stakeholders and define the comprehensive requirements that enable design and implementation of the ISF.

Project activities were initiated by holding the Transparency Workshop: Development of an Information Sharing Framework in December 2011, in Tokai, Japan (Workshop 2011). During Workshop 2011, it was agreed that a step-by-step approach would be taken and that a 'model ISF' would be established with as few conditions as possible. The audience was narrowed down to experts in the four partner organizations (SNL, KINAC, KAERI and the JAEA) and the content to be shared was limited to non-proliferation relevant information. After requirements and the model ISF is established, a demonstration and feedback are to be carried out. In the long term, this model ISF would be expanded to invite other interested organizations and include other information [3]. For example, the ISF could potentially include stakeholders interested in sharing best practices, lessons learned, or technology related to the development of fast reactors.

During Workshop 2011, project partners have used telephone conferences and emails, and thus far discussed essential features of the model ISF, including the goals and needs, stakeholders, types of information to be shared and ways of information sharing. Another transparency workshop was planned to take place in December 2012 in Daejeon, Republic of Korea, where the requirements for the ISF would be further discussed and outlined.

2. REQUIREMENTS FOR ESTABLISHING AN INFORMATION SHARING FRAMEWORK

2.1. Need for requirements

In the past, the importance of transparency of nuclear energy programmes has been repeatedly identified as a key measure for international and regional confidence building. There have been many endeavours for establishing and implementing mechanisms for information sharing; however, thus far, they have not been greatly successful or adopted widely [4]. This largely stems from the inherent challenges of transparency implementation such as its voluntary nature and lack of structured mechanisms. The requirements for the ISF need to address these challenges. Below are two important needs for the requirements.

(1) *Need for maintaining sustainability*. Transparency is, by definition, a voluntary undertaking not derived from compelling obligations. It is therefore a tempting target for cost and human capital cutting, especially during times of scarcity [4]. Therefore, maintaining sustainability has been one of the biggest challenges for the successful implementation of information sharing. In order to address this challenge, the requirements for ISF need to encourage continuous engagement of the relevant parties and maintenance of the framework. It would be important to evaluate the ISF's effectiveness, activity and usability, and to demonstrate its value and significance to the parties, their sponsors, and other potential stakeholders, as a part of routine meetings as well as outreach activities. A separate mechanism to support sustainability is to establish the ISF within an existing agreement, such as the DOE–JAEA agreement, a professional network, such as the Asia Pacific Safeguards Network (APSN) [5], or another setting in which bi- and multilateral parties participate.

(2) *Need for identifying clear steps*. Information sharing has been discussed on various occasions as a means for enhancing nuclear transparency. However, most of the efforts have been devoted to identifying 'what' kind of information is to be shared, but not describing 'how' to share the information. As a result, there are no specific mechanisms for information sharing [6]. Considering this background, requirements are needed to provide clear steps for planning and implementing an ISF so that they can support an infrastructure for 'how' to share the information. If 'how' is clearly identified, parties can see how their concerns are going to be addressed under the ISF, i.e. information security, workload as an information provider, etc. Benefits of the ISF will also be more easily and clearly identified, i.e. types of information that can be received, what receivers can assess from the information, etc. If both concerns and benefits are clear in the planning, it can possibly lower the hurdle for stakeholders to participate in the activities.

On the basis of these needs, project partners are currently making an effort to develop the requirements for an ISF that are to be finalized in the near future. Once the requirements are established, the current project partners will develop, test and demonstrate the model ISF.

2.2. Requirements for the ISF

The primary requirement for the ISF is defined, at this stage, as "implementing the Plan-Do-Check-Adjust (PDCA) cycle[1] for each category of information to be shared. Planning consists of defining the requirements elements, including objectives, audience, scope, content, amount of information to share, frequency of sharing information, information quality, infrastructure and sustainability."

The PDCA cycle is often used to develop a systematic and comprehensive approach for continuous improvement of operations or project management. As the cycle repeats, the performance should be continuously improved, which should contribute to sustainability. The partners engaged in the development of the model ISF adopted this concept of the PDCA cycle so as to provide clear steps for planning and implementing, and encouraging the sustainability of the ISF. Figure 1 shows an explanation of how the PDCA cycle is incorporated in the development and execution of the ISF.

[1] The PDCA cycle is an iterative four step process to plan products and processes, execute the plan, check the results and, finally, to adjust the plan based on the results.

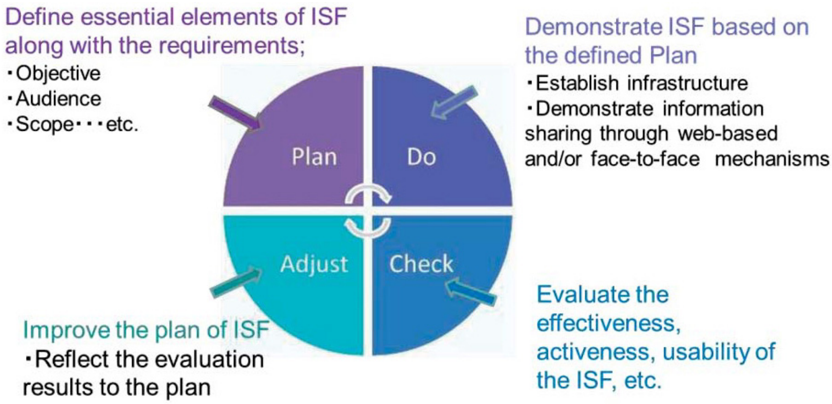

FIG. 1. PDCA cycle for ISF implementation.

Plan: Develop a plan by defining requirements elements of the ISF for each category of information. The next steps of ‘Do, Check and Adjust’ should be included in the plan so that the ISF implementation can automatically follow the PDCA cycle.

Do: Demonstrate the ISF as defined in the plan. This ‘Do’ step consists of two parts: establishing the infrastructure and demonstrating the information sharing.

Check: Evaluate the effectiveness, usability, activity, etc., of the ‘Plan’ and ‘Do’ steps.

Adjust: Reflect on the findings from the ‘Check’ to the ‘Plan’ steps to encourage continuous improvement.

2.3. Requirements elements

Requirements elements for the ISF were identified based on the ‘eight element structure’ of transparency suggested by Baldwin and et al. [4]. These subdivided elements of transparency cover the important aspects of information sharing, including concerns and expectations of involved parties; therefore, they can be a good basis for development of the requirements for the ISF. They include:

(1) *Objectives.* The objective of sharing information should be defined. Considering that the definition of transparency used for the ISF project is to “independently assess the legitimate management of nuclear material”, the objective can be defined by asking the question, “What do information providers expect receivers to assess and what do receivers want to assess with the shared information.”

(2) *Audience (information receivers/providers).* Identify and characterize the audience based on the defined objectives by asking, "In what entity and at what level within the entity should the information receivers and suppliers be?"
(3) *Scope.* Define the boundaries within which the effort will be implemented by focusing on significant needs and concerns of the audience.
(4) *Content*
    (a) Specific information to be shared is selected to achieve the objectives. It will be selected to be of interest to stakeholders and releasable to intended receivers by providers, i.e. information security, intellectual property issues.
    (b) The amount of information to be shared.
    (c) The frequency of sharing information should be also clearly defined as this directly affects the workload and other resources needed for initiating and implementing information sharing.
(5) *Security and credibility.* Security and credibility should be taken into account to address concerns of both information providers and receivers. Major concerns of receivers may be authentication that the information comes from its assumed provider without tampering or impersonation by fraudulent actors; credibility of the information from the provider (for example, assurance that information is not intentionally altered by providers). Providers also need to be confident that information can be transmitted to destined receivers under proper security without intrusions or theft, and information would be appropriately protected after sharing, if necessary. Both providers and receivers need to be careful that sharing information does not increase the chance of misunderstanding and possibly increase mistrust.
(6) *Infrastructure.* Select effective and efficient infrastructure (face-to-face and/or web based [7]) for sharing the information, taking into consideration the limitations of cost and resources, access control, types and security level of information, and audience. Maintenance of the infrastructure should also be taken into account, i.e. how to continuously maintain an understanding of changes in partner organizations while also maintaining equipment, recordkeeping, information sharing tools, etc.
(7) *Sustainability.* As transparency is a voluntary process, it is essential to evaluate the framework and ensure that it is active and viable. For the evaluation, metrics and methodologies need to be defined. Also, it may be a good idea to inform entities/individuals outside the framework about this effort and obtain feedback from them. Outside review can be a basis for peer review and prospects for future possible expansion.

The plan is developed by defining these seven requirements elements. Next, the ISF is implemented as defined, following the PDCA cycle, which will help with continuous improvement and sustainability [8]. Under the current project, these requirements elements are at the very initial stage of development, and they will be refined as the discussion among project partners evolves.

3. EXAMPLE PLANNING BY APPLYING THE REQUIREMENTS

To help understand the requirements and capture an image for planning, Table 1 shows an example plan of the model ISF by defining the requirements elements. Here, safeguards experiences and lessons learned, suggested during Workshop 2011 [3], were used as an example.

TABLE 1. EXAMPLE PLANNING

Elements	Example: **Safeguards experiences and lessons learned**
1. Objectives	• *Improve* R&D planning, safeguards implementation, nuclear material management system by reflecting on other organizations' experiences and lessons learned (information receiver). • *Gain an understanding* of other organizations' safeguards commitments (information provider). • *Facilitate* IAEA safeguards by improving States' capabilities. • *Confidence building* to be promoted among the audience by learning about each other's willingness and commitment to transparency (both receiver/provider).
2. Audience (information receivers/providers)	*Project partners of ISF: SNL, JAEA, KINAC, KAERI* Experts working in the areas of: • Safeguards technology R&D; • Safeguards implementation.
3. Scope	• Non-sensitive, safeguards related information, i.e. published papers, open source information, documents authorized to share through the ISF; • Face-to-face oral presentations and additional explanations on specific topics.

TABLE 1. EXAMPLE PLANNING (cont.)

Elements	Example: **Safeguards experiences and lessons learned**
4. Content	(a) Specific types of information to be shared: Paper list, PPT, and other materials presented at international conferences, i.e. INMM annual meeting, APSN, etc., about: — R&D of NDA equipment and remote monitoring system; — Best practices and lessons learned from R&D activities. (b) Amount: Paper list of INMM[1] annual meeting, APSN[2], etc.. (e.g. 2 page list), PPT presented at recent international conferences (e.g. 1 MB+ 20 files), links to the the related web sites. (c) Frequency: Update accordingly, reviewed annually.
5. Security and credibility	*Web based*: • On-line libraries should be designed while taking into account IP issues. • Web site should be designed and established to consistently follow all the participating organizations' security policies. • Use encryption for data transmission if necessary, take appropriate measures to prevent tampering. *Face-to-face*: • Limit participants to experts who are authorized by their organizations.
6. Infrastructure	*Web based* infrastructure to be developed following the steps below: • Develop web site/on-line library for information sharing. • Upload content to the web site/on-line library e.g. web site: announcement of training, workshops, etc./on-line library: published papers, meeting materials, workshop agendas, etc. • Secure adequate human resources for maintenance of web systems. *Face-to-face*: • Determine an appropriate framework for holding face-to-face meetings (e.g. under inter-organizational agreements, at side meetings of international conferences, using existing frameworks such as APSN). • Hold face-to-face meetings regularly (1~2/year) and share relevant information.

[1] The Institute of Nuclear Materials Management (INMM) is an international technical and professional organization that works to promote the safe handling of nuclear material and the safe practice of nuclear materials management through publications, as well as organized presentations and meetings such as the annual meeting.

[2] The Asia Pacific Safeguards Network (APSN) is an initiative for the region that reflects the commitment of the safeguards community to the peaceful uses of nuclear energy underpinned by effective safeguards implementation.

TABLE 1. EXAMPLE PLANNING (cont.)

Elements	Example: **Safeguards experiences and lessons learned**
7. Sustainability	*Establish metrics* to assess the effectiveness and activity of the framework: • Measure frequency of visits to the library and web site. • Usability, user-friendliness, user-assessment of the web site and on-line library. • The number of face-to-face meetings. • Satisfaction level of the contents for improving the R&D plan, safeguards, implementation, nuclear material management system, etc. • Assess the level of the workload to support the ISF and information sharing. *Implement evaluation* of the framework using the metrics above and: • Self-assessment: Regularly implement self-assessment by the audience in the participating organizations (SNL, KINAC, KAERI, JAEA) through feedback functions of the web site and face-to-face meetings. • Peer review: Implement peer review by inviting selected non-proliferation experts (safeguards R&D and/or safeguards implementation) from outside participating organizations to the web site and to the face-to-face meeting as observers to obtain feedback from them. *Assess context* • Inform and solicit feedback about ISF project from non-proliferation experts outside the framework by making presentations at conferences such as the INMM where a larger number of non-proliferation experts gather, or meetings of APSN, a professional organization with a similar objective of exchanging safeguards experiences to improve the level of performance and expertise, or other non-proliferation communities. • Explore the possibilities of cooperating with other communities, inviting other organizations/individuals to the framework, or sharing other types of information. *Improve the plan for the framework* • Reflect on findings from self-assessment and peer review, outreach and discussions about possible expansion to plan the framework and encourage continuous improvement.

By defining these requirements elements, the countermeasures needed to address concerns and the benefits of sharing safeguards experiences and lessons learned are identified and become apparent to stakeholders, which can encourage their participation in the ISF. It should also be noted that each category of information should be examined independently because requirements elements might be different, depending on the information to be shared.

4. OBSERVATIONS

Discussion of the requirements for the ISF is still under way; however, project partners are expected to finalize them in the near future. The next step will be to identify several categories of information for sharing. Then planning will be developed for each category of information following the procedure shown in the example plan. Infrastructure for a model ISF will be established and information sharing will commence as defined in the plan. After a certain period of implementation of the model ISF, activity will be evaluated as to its effectiveness, usability and other important features, and the findings will be reflected in the plan. By following this process, the PDCA cycle can be repeated for each category of information and demonstration of the model ISF will be implemented in a systematic manner.

If the model ISF demonstration is successful, the approach could be expanded in the longer term to invite new partners and include other categories of information. During this stage of expansion, collaboration with larger, international, multilateral frameworks, i.e. APSN, will also be taken into consideration.

Ultimately, it is expected that a comprehensive ISF or series of ISFs for the Asia Pacific region should be established to complement and reinforce IAEA safeguards and promote international and regional confidence building. In fact, a similar information sharing scheme could be considered for use by organizations involved in the development of fast reactors. Organizations responsible for fast reactor planning and implementation would benefit by sharing non-sensitive information such as best practices and lessons learned, while concerns about nuclear safety, security and safeguards could be addressed through various information sharing mechanisms, such as meetings, on-site tours, web sites, etc.

5. CONCLUSION

This paper introduced the current status of the requirements development for an ISF. The requirements are currently defined to establish a plan for each category of information to be shared by defining seven requirements elements, and executing the plan so that implementation of ISF follows the continuous improvement PDCA cycle. The ISF requirements are still under discussion; however, the joint project of establishing “An Information Sharing Framework for Regional Non-proliferation Cooperation” has made substantial progress. This joint project continues steps towards establishing, demonstrating and evaluating an ISF, and will then consider the next steps. Ultimately, this ISF approach is expected to complement and reinforce IAEA safeguards and promote international and regional confidence building.

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IAEA ACTIVITIES IN THE AREA OF FAST REACTORS AND RELATED FUELS AND FUEL CYCLES

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Abstract

In the actual context of growing energy needs on one side and concerns for the environment on the other, it is generally recognized that innovative fast reactors and fuel cycle concepts will be able to provide a relevant contribution to future energy needs, if the research and technology developments create the conditions to clearly satisfy the criteria of economic competitiveness, stringent safety requirements, sustainable development and public acceptability. For more than 45 years, the IAEA has been accompanying and supporting the development and deployment of the fast reactor technology, serving the interested Member States as a major forum for fast reactor information exchange and collaborative research and technology development. In particular, since 1967 the keystone of the IAEA's efforts in this field is represented by the Technical Working Group on Fast Reactors (TWG-FR), which is a group of experts tasked to provide advice and support programme implementation, reflecting a global network of excellence and expertise in the area 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 (TWG-NFCO), the Department of Nuclear Sciences and Applications, the Department of Nuclear Safety and Security and, last but not least, the International Project on Innovative Nuclear Reactors and the Fuel Cycle (INPRO). Among the broad spectrum of IAEA activities in this field, the coordinated research projects (CRPs) represent the major tool to enhance Member States' knowledge and technical capabilities in the different fields of the fast reactor technology, as well as to promote international cooperation and sharing of knowledge. With regard to the last purpose, the IAEA regularly organizes technical meetings and conferences to discuss the main technology challenges facing the deployment of fast reactors and advanced fuel cycles, to present the results of R&D programmes and to propose future activities to be implemented within the IAEA programmes. A further important contribution is represented by the publication of scientific and technical reports on different topics of fast reactor technology. This paper aims to present the main IAEA activities in the field of fast reactors and related fuel cycles, including advanced fuels and materials. It presents and discusses the recently completed, ongoing and planned CRPs in the field. Additionally, the paper gives a comprehensive overview of the most relevant technical meetings organized by the IAEA, as well as of the recent publications on fast reactor technology and related fuel and fuel cycle concepts.

1. INTRODUCTION: STATUS OF FAST REACTORS AND RELATED FUEL CYCLES DEVELOPMENT

The potentialities of fast spectrum nuclear reactors have been recognized from the very inception of nuclear energy, dating back to the 1950s. By the achievable breeding ratio and the multi-recycling of the fissile materials obtained from the spent fuel, fast reactors allow full utilization of the energy potential of the natural resources (uranium and thorium), thus drastically enhancing the sustainability of nuclear power in terms of resource preservation and management of high level and long lived radioactive wastes. However, the complex technology intrinsically required by fast reactors has not allowed the same successful deployment of thermal reactors.

Significant R&D programmes have been pursued in the past worldwide, bringing the knowledge on fast reactors and associated fuel cycles technology to a high level of maturity. As of today, several fast reactor construction projects are currently ongoing; among them, examples of current sodium cooled fast reactors are the BN-600 in the Russian Federation, the China Experimental Fast Reactor which was connected to the grid in July 2011, the Russian BN-800 and the Prototype Fast Breeder Reactor in India, both under construction.

In the actual context of growing energy needs and concern for the environment, a successful large scale deployment of fast reactors is reasonably achievable only if the research and technology developments create the conditions to exploit the full potential of the fast neutron systems and related closed fuel cycles, and if the criteria of economic competitiveness, stringent safety requirements, sustainable development and public acceptability are clearly satisfied. It is therefore of paramount importance to gain understanding and to assess different design options and related safety characteristics, based on past knowledge and experience, as well as on new scientific and technological research efforts.

In this context, the IAEA, which has been accompanying and supporting the development of fast reactors and related fuel cycles technology for almost 50 years, plays a prominent role, representing for the interested Member States the major fulcrum for scientific and technical cooperation in this field.

2. OBJECTIVES AND FRAMEWORK OF IAEA ACTIVITIES ON FAST REACTORS AND RELATED FUEL CYCLES

The IAEA, coherently with its statutory role and objectives, provides support for technology development of advanced fast reactors and associated fuel cycles, assisting Member States operating fast reactors or developing new reactors and fuel cycle processes and facilities, with the main aim of catalysing innovation and technology advance in that field. To reach this goal, the IAEA establishes cooperative research activities, promotes information exchange through the organization and conduct of international conferences, workshops, meetings and training schools, and publishes technical and scientific material on different aspects of fast reactor systems and advanced materials, fuels and fuel cycles.

The IAEA's activities in the field are mainly implemented within the framework of various Technical Working Groups (TWGs), specifically the TWG-FR, the TWG-NFCO and the TWG on Fuel Performance Technology (TWG-FPT).

The TWG-FR of the IAEA Department of Nuclear Energy was established by the former IAEA Director General, Sigvard Eklund, who was in charge from 1961 until 1982, in response to the interest of several Member States in developing experimental fast neutron reactors. Today, the TWG-FR consists of representatives from 20 Member States and three international organizations (Belarus, Brazil, China, France, Germany, India, Italy, Japan, Kazakhstan, Republic of Korea, Netherlands, Russian Federation, Sweden, Switzerland, Ukraine, United Kingdom, United States of America, OECD/NEA, European Commission, Argentina (observer), Belgium (observer), Spain (observer)), providing advice and support programme implementation, reflecting a global network of excellence and expertise in the area of advanced technologies and R&D on fast reactors and subcritical hybrid systems for energy production and utilization/transmutation of long lived nuclides [1]. The TWG-FR assists in defining and carrying out the IAEA's activities in the field of technology development for fast reactors, in accordance with its Statute, and ensuring that all the activities are in line with the expressed needs of Member States. It promotes in-depth scientific and technical exchange of information on national and multinational programmes and new developments and experience, with the goal of identifying and reviewing problems of importance and of stimulating and facilitating cooperation, development and the practical application of fast reactors and subcritical hybrid systems.

The TWG-FR Member State's representatives meet every year in the annual meeting of the TWG. The 45th TWG-FR meeting was held at Argonne National Laboratory, Chicago, on 20–22 June 2012 [2]. Besides the exchange of information on national and international programmes on fast reactors, an important outcome of the meeting was a list of priorities for new fast reactor related activities to be carried out in the next years, e.g. new coordinated research projects (CRPs), technical meetings and workshops, reports, handbooks and guidelines. Participants are shown in Fig. 1.

FIG. 1. Participants at the 45th annual meeting of the TWG-FR.

The TWG-NFCO was established to support the IAEA's efforts to assist its Member States in their nuclear fuel cycle activities by providing an international cooperation tool [3]. The TWG-NFCO is a group formed from the merger of the Regular Advisory Group on Spent Fuel Management and the International Working Group on Nuclear Fuel Cycle Options. The TWG-NFCO mainly deals with:

- Back end fuel cycle policies and strategies to which first priority is given in the IAEA's medium term strategy in regard to nuclear energy;
- Spent fuel management;
- Fuel cycle options and relevant issues;
- Nuclear material management.

The TWG-FPT was set up in 1976 and now consists of experts from 25 Member States and two international organizations [4]. The TWG-FPT focuses its work on status and trends in nuclear power reactor fuel performance and technology. It covers nuclear core material's R&D; fuel design, manufacturing and utilization; coolant chemistry; fuel performance analysis and quality assurance issues.

Besides the TWGs, an important contribution to support innovative fast reactors and advanced fuel cycle development is provided by the IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO), which was established in 2000 to help ensure that nuclear energy is available to contribute to meeting the energy needs of the 21st century in a sustainable manner. Some INPRO activities in the field are presented in other papers and contributions at this conference (e.g. Ref. [5]).

It is important to underline that the IAEA cooperates with other relevant fast reactor and related fuel cycle initiatives, implemented within the framework of international programmes such as the Generation IV International Forum (GIF), the Nuclear Energy Agency (OECD/NEA), the Euratom programmes and projects and the European Sustainable Nuclear Industrial Initiative (ESNII).

3. INTERNATIONAL CONFERENCES, WORKSHOPS, SEMINARS AND TECHNICAL MEETINGS

The IAEA regularly organizes conferences, workshops, seminars and technical meetings in the field of fast reactors and related fuels and fuel cycles.

The International Conference on Fast Reactors and Related Fuel Cycles, which is held every four years, represents the most important event on fast reactors and related fuel cycle technology. The last conference (FR09) was held in Japan in 2009, and the proceedings have been recently published by the IAEA (Fig. 2(a), [6]). Key topics of the Conference were national and international programmes on fast reactors and related fuel cycles, innovative fast reactor development, coolant technologies, fast reactor component design, safety, materials, fuels and fuel cycles, experiments and simulation, experience in operation and decommissioning, and knowledge management.

With similar purposes, the FR13 conference [7] is aimed at exchanging information on national and international programmes, and more generally on new developments and experiences in the field of fast reactors and related fuel cycles. It is worth mentioning that this event includes a very interesting young generation event, dedicated to young professionals involved in fast reactor and fuel cycle projects.

3.1. Workshops and technical meetings on new fast reactors technology

A significant number of technical meetings on specific scientific and technical aspects of fast reactors and related fuel cycles were recently conducted by the IAEA with the aim of fostering exchange of information on recent technology advances arising from national and international research programmes, as well as identifying gaps to be covered with new research activities [8]. A list of IAEA

a) b)

FIG. 2. (a) The Proceedings of the FR09 international conference. (b) The poster of the FR13 international conference.

Technical Meetings held in 2011 and 2012 focused on different topics concerning the development of new fast reactors is as follows:

- Fast Reactors Deployment Issues, 14–16 February 2011, Vienna, Austria
- Fast Reactor Physics and Technology, 14–18 November 2011, IGCAR, India
- Fast Reactor In-service Inspection and Repair: Status and Innovative Solutions, 19–20 December 2011, Vienna, Austria
- Innovative Heat Exchanger Designs for Fast Reactors, 21–22 December 2011, Vienna, Austria
- Innovative Fast Reactor Designs with Enhanced Negative Reactivity Feedback Features, 27–29 February 2012, Vienna, Austria
- Identify Innovative Fast Neutron Systems Development Gaps, 29 February – 2 March 2012, Vienna, Austria

The IAEA Education and Training Seminar/Workshop on Sodium Cooled Fast Reactor Science and Technology was held in Bariloche, Argentina, on 21–25 February 2011; following the successful results of this event, Argentina's CNEA hosted, on 1–5 October 2012, a second educational/training Workshop/Seminar on Fast Reactor Science and Technology, addressing this time all the different types of fast reactor being developed worldwide. The event was a very fruitful initiative to promote education and training activities for young scientists and engineers involved in fast reactor projects, who had the occasion to learn about nuclear power development and deployment senarios, fast neutron systems, and specific technical aspects and issues of the different fast reactor concepts being developed worldwide.

The IAEA also organizes every two years at the International Centre for Theoretical Physics in Trieste (Italy) a School on Physics, Technology and Applications of Innovative Fast Neutron Systems and Related Fuel Cycles.

3.2. Workshops and technical meetings on fast reactors fuels, materials and fuel cycles

A technical meeting on design, manufacturing and irradiation behaviour of fast reactor fuels was held on 30 May – 3 June 2011 in the Institute of Physics and Power Engineering (IPPE), Russian Federation.

The effective cross-cutting international coordination of nuclear science and technology development programme is implemented through a number of different activities. Since 2009, the IAEA, jointly with the EC, regularly implements international topical meetings on Advanced Fission and Fusion Reactor Systems. This particular activity primarily helps to foster interaction between scientist and engineers in the area of nuclear fusion and fission technologies in order to accelerate and strengthen further developments and capacity building in interested Member States. Special focus is given to the functional and structural materials and their compatibility with the coolant, long term improved performance (beyond end-of-life limit), and the study and analysis of various degradation mechanisms, as well as qualification of new materials. A list of the main topics addressed during the topical meeting is reported below and the full proceedings were published as a special issue of a peer reviewed journal [9]:

- Key operational conditions and material parameters of selected reactor designs;
- Microstructures and mechanical properties of new nuclear structural materials;
- Ongoing challenges in radiation damage phenomena and modelling;

— Coolant compatibility issues;
— Pathways to the development and qualification of new structural materials;
— Advanced microstructural probing methods.

3.3. Workshops and technical meetings on fast reactor safety

Several initiatives recently launched by the IAEA were devoted to discussing the safety aspects of fast reactors, in particular to analyse the impact of the Fukushima event on the construction and operation of current fast reactors and on the design of new systems. The most relevant ones have been the following:

— A series of Joint GIF–IAEA Workshops on Safety Aspects of Sodium Cooled Fast Reactors;
— An International Workshop on Prevention and Mitigation of Severe Accidents in Sodium Cooled Fast Reactors, jointly organized in June 2012 by the IAEA and the Japan Atomic Energy Agency;
— The GIF–IAEA Workshop on Safety Design Criteria for Sodium Cooled Fast Reactors, whose main outcomes will be presented at this Conference during the panel devoted to safety design criteria for fast reactors.

Another relevant initiative in this area was the technical meeting, Impact of the Fukushima Event on Current and Future Fast Reactor Designs, which was hosted by the Helmholtz-Zentrum Dresden-Rossendorf research centre in March 2012. The experts participating at the meeting recognized several issues raised by the Fukushima accident, and identified areas of common interest to be investigated by IAEA CRPs.

4. IAEA COORDINATED RESEARCH PROGRAMME ON FAST REACTORS AND RELATED FUEL CYCLES

The main IAEA framework for establishing coordinated research activities is through the implementation of CRPs [10] that also represent an important opportunity to enhance international cooperation as well as to share information and technical know-how. The following sections give an overview of the recently completed, ongoing and planned CRPs focused on the most challenging technical areas of fast reactors and related fuel cycle development.

4.1. Projects in support of the design of advanced fast neutron systems

A CRP recently completed, Analyses of and Lessons Learned from Operational Experience with Fast Reactor Equipment and Systems, which contributed to the preservation of the feedback from the commissioning, operation and decommissioning of experimental and power sodium cooled fast reactors. This objective was pursued by the retrieval and assessment of all the relevant technical documentation and information on the operation of fast reactors and will contribute to feeding the IAEA Fast Reactor Knowledge Organization System.

Among the CRPs recently completed, it is worth mentioning Benchmark Analyses of Sodium Natural Convection in the Upper Plenum of the Monju Reactor Vessel, whose objective is to improve Member States' knowledge in the field of fast reactors in-vessel sodium thermohydraulic phenomena. In particular, the CRP addresses the natural convection behaviour of the coolant in the reactor vessel of a loop type sodium cooled reactor. The CRP participants have performed a set of benchmark numerical simulations aimed at reproducing the thermal stratification measured in the upper plenum of the Monju reactor vessel, after a plant strip test conducted in December 1995 with the reactor at 45% thermal power level, simulating an abnormality in the condenser as a triggering event. The CRP has allowed the validation of various multi-dimensional fluid dynamics codes in use in Member States through simulation of sodium cooled fast reactor outlet plenum temperature distributions and comparison with experimental data. It has also identified weaknesses in current methodologies (e.g. with regard to turbulence models, reactivity feedback models) as well as new R&D needs to resolve the open issues.

A second CRP recently completed is Control Rod Withdrawal and Sodium Natural Circulation Tests Performed during the Phenix End-of-life Experiments. The objective of this CRP is to perform several benchmark analyses on the final sets of experiments carried out in the French prototype power fast breeder reactor Phenix, before its definitive shutdown. The CRP has aimed at the improvement of capabilities in sodium cooled reactor simulation through code verification and validation, with particular emphasis on temperature and power distribution calculations, and the analysis of sodium natural circulation phenomena. The two Phenix end-of-life tests which have been investigated are the control rod withdrawal test and the sodium natural circulation test. The control rod withdrawal test was performed in both the static and dynamic modes; the comparison of the results allows sensitivity analyses of the two measurement methods to be performed and provides the basis for improving the uncertainty in the determination of power distributions. As regards the sodium natural circulation test, the objective is twofold: the study of the sodium natural circulation in the primary circuit as well the determination of the efficiency of natural convection phenomena in the primary circuit, and the qualification of the system codes used to simulate natural convection phenomena.

With analogous purposes to the previous projects, a CRP on Benchmark Analyses of an EBR-II Shutdown Heat Removal Test has been recently established by the IAEA. The CRP addresses shutdown heat removal tests executed in the Experimental fast Breeder Reactor (EBR-II) within the framework of the US Integral Fast Reactor Development and Demonstration Program. Even this CRP is intended to improve the participants' simulation capabilities in the various fields of research and design of sodium cooled fast reactors through data and code validation and qualification. The scope of the CRP is twofold: firstly, validation of the state of the art liquid metal cooled fast reactor codes and data used in neutronics, thermohydraulics and safety analyses, and secondly, training of the next generation of fast reactor analysts through international benchmark exercises.

Another initiative in the field of fast reactors innovative technologies, carried out in the framework of the IAEA INPRO project, is the collaborative project Integrated Approach for the Modelling of Safety Grade Decay Heat Removal System for Liquid Metal Reactors (DHR). The activity deals with the inter-comparison of results of a candidate robust safety grade DHRS for liquid metal cooled reactors.

The IAEA is also involved in a number of projects aimed at supporting Member States in developing ADS technology. A CRP on Analytical and Experimental Benchmark Analyses of Accelerator Driven Systems, which was concluded in 2010, aimed at improving current understanding of the coupling of an external neutron source (e.g. a spallation source in the case of the ADS) with a multiplicative subcritical core. In a previous IAEA CRP on Use of Th Based Fuel Cycle in ADS to Incinerate Pu and to Reduce Long Lived Waste Toxicities, reactor physics benchmark calculations on ADS with fixed external neutron sources were performed.

A CRP on Sodium Properties and Design and Safe Operation of Experimental Facilities in Support of the Development and deployment of Sodium Cooled Fast Reactors (SFRs) — NAPRO is to be launched in 2013, in order to address the need for standardization of sodium's physical, physicochemical and thermodynamic properties, the main rules for design, construction and operation of sodium experimental facilities, as well as good practices and safety guidelines for sodium experiments.

4.2. Projects on fast reactors fuels and materials

The CRP on Accelerator Simulation and Theoretical Modelling of Radiation Effects addresses irradiation behaviour (including very high doses that are not currently achievable in neutron irradiation experiments) of steels used and planned to be used for fast reactor fuel cladding and other critical structural components of advanced nuclear reactors. The final report of this CRP is in preparation and will be published in 2013.

Another ongoing CRP on Benchmarking of Structural Materials Pre-selected for Advanced Nuclear Reactors focuses on the development and characterization of ferritic and ferritic-martensitic oxide dispersion strengthened steels for fusion and fission applications. This particular project significantly contributes to the R&D and pre-qualification phase of oxide dispersion strengthened materials for further use in advanced fast reactors, in particular for fuel cladding. The CRP will be finished in 2014 and the main results will be published in 2015. A more detailed overview of the activities on R&D and the testing of structural materials for advanced nuclear energy power reactors is summarized in Ref. [11].

4.3. Projects on advanced fuel cycle technologies and scenarios

The IAEA has given a high priority to projects on advanced partitioning processes as these processes, based on either aqueous or pyro, play an important role for the successful deployment and expansion of nuclear power on a long term basis. The objectives of these processes include reuse of separated fissile materials from spent nuclear fuels to obtain energy, enhancement of resource utilization, reduction in the disposal of toxic radionuclides and improvement of the long term performance of geological repositories.

Many Member States are involved in the development of an advanced nuclear fuel cycle that could effectively incorporate actinide recycling involving advanced partitioning processes, based on either aqueous or pyro, to reduce inventories of plutonium and minor actinides. A Technical Meeting on Advanced Partitioning Processes was held in Vienna in June 2011. A draft document has been prepared based on the inputs received during the meeting and also from the Member States actively involved in the development of partitioning processes. The IAEA has also planned to organize a Technical Meeting on Advanced Recycle Technologies in 2013 with the objective of providing a common platform for the scientists and engineers working in the areas of reprocessing of spent nuclear fuels and manufacturing of advanced fuels for fast reactors in order to bridge the technological gap between them.

5. TECHNICAL PUBLICATIONS

One of the most important outputs of IAEA work in the area of fast reactors and related fuel cycles is represented by the publication of technical documents, more specifically IAEA-TECDOC series and Nuclear Energy Series. Examples of recent publications in the field of fast reactors and fuel cycle technology are shown in Fig. 3.

The status report on fast reactors, which is intended to provide comprehensive and detailed information on the technology of fast neutron reactors, is one of the most important publications of the IAEA in this area. The last report, Status of Liquid Metal Cooled Fast Breeder Reactors, was published by the IAEA in 1985. In 2006, recognizing the need for an updated version, the IAEA, in collaboration with the Member States of the TWG-FR, started work on the new report, Status of Fast Reactor Research and Technology Development, which was recently published by the IAEA [12].

A recent IAEA publication on fast reactors technology is the IAEA Nuclear Energy Series No. NP-T-1.6, Liquid Metal Coolants for Fast Reactors Cooled By Sodium, Lead and Lead-bismuth Eutectic [13], which provides a comprehensive summary of the status of the liquid metal coolant technology, a challenging issue for the development of fast reactors.

A list of other relevant IAEA reports under publication or preparation in the field of fast reactors is provided below:

- Accelerator Driven Systems — Energy Generation and Transmutation of Nuclear Waste. Status Report;

FIG. 3. Examples of recent IAEA publications in the area of fast reactors and related fuel cycles.

- Design Features and Operating Experiences of Experimental Fast Reactors;
- BN-600 Hybrid Core Benchmark Analyses, Results from a Coordinated Research Project on Updated Codes and Methods to Reduce the Calculational Uncertainties of the LMFR Reactivity Effects;
- Final reports of the CRP on Control Rod Withdrawal and Natural Circulation Tests Performed during the PHENIX End-of-life Experiments;
- Final report of the CPR on Benchmark Analyses of Sodium Natural Convection in the Upper Plenum of the MONJU Reactor Vessel;
- Final report of the CRP on Analyses of and Lessons Learned from the Operational Experience with Fast Reactor Equipment and systems.

In the area of fast reactors fuels and materials, the recently published IAEA Nuclear Energy Series No. NF-T-4.3, Structural Materials for Liquid Metal Cooled Fast Reactor Fuel Assemblies — Operational Behaviour [14] summarizes findings of several consultancies and technical meetings. The report complements the 2011 publication Status and Trends of Nuclear Fuels Technology for Sodium Cooled Fast Reactors [15], which covers status and trends of fuels technology for sodium cooled fast reactors, highlighting the manufacturing processes, out-of-pile properties and irradiation behaviour of MOX, MC, MN and metallic U-Zr and U-Pu-Zr fuels. The information compiled in these books is a valuable resource for materials scientists and engineers involved in the development of fuels for fast reactors. Another publication on Status of Developments in the Back End of the Fast Reactor Fuel Cycle [16] highlights emerging innovations and R&D needs for the back end of the fast reactor fuel cycle with emphasis on reprocessing of U and Pu based ceramic and metallic fuels.

During the past decade, an urgent need has arisen for the development of new advanced materials for new nuclear reactors (both fusion and fission concepts). The research reactors offer unique and dedicated services which are needed for the testing and qualification of new structural materials. Today's multipurpose test research reactors are primarily used for irradiation services, neutron radiography, as well as beam research and material characterization. The recently published IAEA-TECDOC on Research Reactor Application for Materials under High Neutron Fluence [17] gives a good overview of practical information and related infrastructure. Specific emphasis is given to the following areas;

— Development and operation of irradiation facilities for testing, characterization and qualification of new structural materials;
— Study of radiation damage mechanisms of such materials at high doses, dose rates, presures and temperatures;

— Sharing of best practice on advanced nuclear technologies by international collaboration and regional networking;
— Fostering of know-how dissemination and training activities.

Besides the scientific and research activities and commercial applications, the research reactors are also used extensively for educational training activities for scientists and nuclear engineers.

6. CONCLUSIONS

Coherently with its Statute, the IAEA actively supports the development of fast reactors and related fuel cycle technology, recognizing its potentiality to meet — in a sustainable and competitive manner — the future world energy needs.

The IAEA mainly serves interested Member States as a major fulcrum for information exchange and technical cooperation. The International Conference on Fast Reactors and Related Fuel Cycles is the main event organized by the IAEA in this field. A significant number of dedicated topical and technical meetings, seminars, workshops and schools were also recently conducted in order to discuss and address technical challenges, innovations and advances in fast reactor technology and associated fuel cycles. This helps to identify technical gaps and new R&D needs, as well as to support the education and training and knowledge preservation in the area of fast reactor technology.

The IAEA's CRPs represent the main framework to perform collaborative research activities among interested Member States. A significant number of CRPs have been performing in the past years, with the aim of helping Member States to improve knowledge in the most significant and challenging scientific and technical aspects of fast reactors and related fuel cycle development, such as, innovative designs for new fast neutron systems, development of advanced fuels, fuel cycle options analysis and safety enhancement.

An important outcome of IAEA activities in this area is represented by the publication of technical and scientific books aimed at supporting scientists and engineers involved in fast reactor and related fuel cycle activities.

It is also worth mentioning that the IAEA is strongly involved in activities devoted to the preservation of the experience and knowledge gained during past decades through the design, construction and operation of experimental, prototype and power fast reactors and the related fuel cycle facilities. More specifically, the IAEA Department of Nuclear Energy's TWG-FR, the International Nuclear Information System and the Nuclear Knowledge Management Section serve as a place for nuclear knowledge accumulation, supporting and coordinating data retrieval and interpretation efforts in the Member States. The main outcomes of these activities are presented at this conference within the track devoted to skill capabilities, professional development and knowledge management [18].

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