Status of Fast Reactor Research and Technology Development, Design, Construction, Operation and Decommissioning

A. Introduction

A.1. Why fast reactors?

When neutrons are generated during the fission process in a reactor they have high energy and are moving fast. Reactors that operate using these fast neutrons are called fast reactors. When fast neutrons hit ^{238}U atoms there is a high likelihood that they are absorbed and that a new atom (after several decays) is produced, ^{239}Pu, which is a new fissile material. This process is called breeding if more fissile material is produced during the operation of the reactor than is consumed for the production of energy.

Thermal reactors, which are the most common, use a different approach. The neutrons are first slowed down by a moderator, either water or graphite, in the reactor core. When the fast neutrons hit atoms or molecules of the moderator, their speed is reduced. Slow neutrons have a high probability of being absorbed by ^{235}U atoms which then leads to fission. Reactors that operate on the basis of slower neutrons are called thermal reactors, which constitute – with few exceptions – all reactors in operation today.

Fast reactors need a high concentration of fissile material to sustain a chain reaction. There are thus strong incentives to extract all fissile material left in the spent fuel and recycle it in new fuel elements. This is called 'closing the fuel cycle'.

Thermal reactors, which mostly use water as both a moderator and coolant and have been used on a large scale, have benefited from major improvements in reliability and economics gained from operating experience.

Nearly all operating nuclear power plants use thermal reactors which can utilize only a small fraction of the energy in uranium. They use mainly the isotope ^{235}U, which is only about 0.7% of natural uranium. The world's identified conventional uranium resources of about 5.5 million tonnes are adequate to fuel today's reactors for roughly another 100 years.

However, fast breeder reactors can use effectively all the energy in uranium by converting the fertile isotope ^{238}U, which is 99.3% of natural uranium, into the fissile isotope ^{239}Pu. Thermal reactors can also convert fertile to fissile material, but breeders can do it in a way that produces more fissile material than is consumed. With fuel reprocessing to retrieve fissile material from the spent fuel, the result is an increase in the energy potential of natural uranium by a factor of about 60.

Finally, the fast neutrons in fast reactors make it possible to use or transmute certain isotopes that cannot be used in thermal reactors and thus normally become part of thermal reactors' waste burden. These are trans-uranium nuclides and some long-lived fission products. By eliminating these isotopes, fast reactors can contribute to reducing the environmental burden of spent fuel, which further enhances the long-term sustainability of nuclear energy. In recognition of the fast reactor's importance to the sustainability of nuclear power, there is currently renewed worldwide interest in fast reactor technology development.

A.2. A short history of fast reactor development

Fast reactor research and technology development programmes started in a number of countries in the 1940s and early 1950s. The USA was first to construct an operational fast reactor, with Clementine becoming critical in 1946. The first kilowatt-hours of nuclear electricity in history were produced in December 1951 by a fast reactor, the EBR-I in Idaho[1], after which the US programme continued with basic R&D and construction of fast reactors of increasing power (EBR-II, FERMI and FFTF)[2]. At essentially the same time, the USSR (BR-10, BOR-60), the UK (DFR), and France (RAPSODIE) also began development programmes and built their own experimental fast reactors. A few years later, Germany and Japan started national development programmes and constructed experimental fast reactors, JOYO and KNK, respectively. Thereafter, the pace of fast reactor development picked up steadily until most programmes reached a peak around 1980.

At this point experimental reactors were operating in many countries, providing R&D tools (mainly as irradiation facilities) for various prototype and commercial sized fast reactor development programmes, e.g. Phénix and Superphénix in France, SNR-300 in Germany, MONJU in Japan, PFR in the UK, CRBR in the USA, as well as BN-350 and BN-600 in the USSR. While interest was increasing in developing countries, the next ten years saw a gradual decline in fast reactor activities in most developed countries. By 1994, the US government had decided to cancel the CRBR and shut down the FFTF and EBR-II. In France, Superphénix was shut down at the end of 1998; SNR-300 in Germany was completed but not put into operation; and KNK-II was permanently shut down in 1991. In the UK, PFR was shut down in 1994, as was BN-350 in Kazakhstan in 1998. At the time, there was simply no compelling need for fast breeder reactors, so most development programmes either proceeded at a reduced scale or stopped. Today, however, renewed interest in nuclear power has heightened awareness of the medium and long term benefits of fast reactors with a closed fuel cycle.

B. Challenges for fast reactor development

The most important challenges for fast reactors are in the areas of cost competitiveness, enhanced safety, non-proliferation and public acceptance. With the exception of public acceptance challenges, these translate into technology development challenges, i.e. the development of innovative reactor, fuel and fuel cycle (reprocessing and fabrication) technology. Some examples are briefly summarized below.

Cost competitiveness can be achieved through simplification, series construction, extension of reactor lifetimes, increased thermodynamic efficiency, reduction of component structural requirements and increased component reliability. Many of these are related to fast reactor core, fuel, coolant and component design. Simplification can be achieved, for example, by eliminating the intermediate heat transfer system. R&D to this end focuses on, on the one hand, gas cooled fast reactors and gas turbines and, on the other, lead cooled fast reactors. Using helium or lead as a coolant avoids two problems with sodium (the coolant in most fast reactor designs), which are the production in the core of the

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[1] The water-cooled, graphite-moderated, 30 MW(t) (5 MW(e)) AM-1 reactor in Obninsk, USSR, delivered the first nuclear electricity to the grid in 1954.

[2] EBR = Experimental Breeder Reactor, FFTF = Fast Flux Test Facility, DFR = Dounreay Fast Reactor, PFR = Prototype Fast Reactor, CRBR = Clinch River Breeder Reactor.

activation product sodium-24 and the fact that sodium reacts violently with water. These problems necessitate an intermediate sodium-sodium circuit to separate the primary sodium from the steam generator. In contrast, helium is not activated in the core, and neither helium nor lead reacts chemically with water. Thus the use of helium or lead as a coolant allows simplified fast reactor designs without intermediate heat removal systems.

Lead cooled loop type fast reactors can also be simplified relative to sodium cooled reactors since lead, unlike sodium, does not react with water. This simplifies steam generator design in lead cooled fast reactors. In helium cooled fast reactors, the steam generator can be eliminated altogether, which reduces costs, by using helium turbines.

R&D includes the development of high-burnup fuels and of advanced structural materials. For high-burnup fuels, the most limiting factor is the cladding material. Currently, stainless steel claddings are state-of-the-art and have made it possible to achieve burnup values as high as 200 GWd/t_{HM}. However, new oxide dispersion strengthened (ODS) steels with increased high temperature oxidation characteristics are being developed. These would both allow longer reactor cycles and lifetimes, and increase thermodynamic efficiency. Efficiency could be further increased in lead or helium cooled fast reactors, which allow higher outlet temperatures than in sodium cooled reactors. In addition to their contribution to cost reductions, long-life cores can contribute to proliferation resistance by reducing, perhaps to nearly zero, the number of times the reactor needs to be opened up for refuelling.

R&D to improve fast reactor safety focuses on reducing the positive reactivity effect due to loss of coolant (known as the void effect). While the neutronics sensitivity of sodium cooled cores to coolant loss can be reduced by various design measures, lead and lead-bismuth eutectic cooled fast reactors offer inherent characteristics that ensure smaller positive void reactivity effects. Specifically, they do not react chemically with water, and they have higher boiling temperatures, 1743°C and 1670°C for lead and lead-bismuth eutectic, respectively, compared to 880°C for sodium.

C. Current status: fast reactor research and technology development, design, construction, operation and decommissioning

C.1. National fast reactor programmes

In **China,** the 25 MW(e) sodium cooled, pool type Chinese Experimental Fast Reactor (CEFR) is under construction, with first criticality foreseen for mid-2009 and grid connection in mid-2010.

CEFR (Figure VI-1) is a 65 MW(t) sodium cooled, pool type experimental fast reactor fuelled with mixed uranium-plutonium oxide. The fuel cladding and reactor block structural material is Cr-Ni austenitic stainless steel. CEFR has two main pumps and two loops for each of the primary, secondary and tertiary circuits. The CEFR block is composed of the main vessel and the guard vessel supported from the bottom on the floor of the reactor pit, which has a diameter of 10 m and a height of 12 m. The reactor core and its support structure are supported on lower internal structures, while the two main pumps and four intermediate heat exchangers are supported on upper internal structures.

The next two stages in the Chinese fast reactor development programme consist of the construction of a 600 MW(e) prototype fast reactor (CPFR), for which design work started in 2005, and a 1000-1500 MW(e) demonstration fast reactor (CDFR).

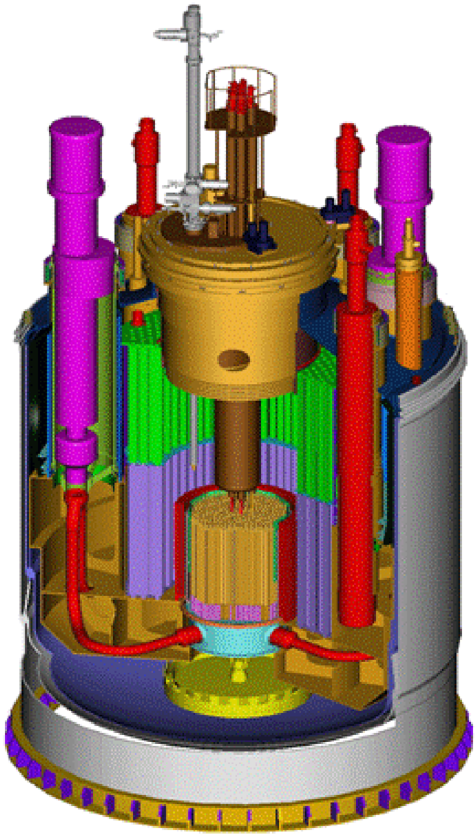

FIG. VI-1. CEFR block scheme.

France has built and operated the 238 MW(e) prototype fast breeder reactor Phénix and the industrial size 1200 MW(e) demonstration fast breeder reactor Superphénix. Both reactors have mixed oxide fuel in the core, are sodium cooled and have a loop type primary system. Phénix became critical in 1973. It is still in operation and has provided operating experience with a complete prototype fast breeder reactor power station. In 2006 Phénix's annual availability and load factors were 78% and 56%, respectively. Two more operating cycles (representing about 250 equivalent full power days (EFPD)) are planned in Phénix before the reactor is finally shut down in 2009. These will allow completion of irradiation tests in support of France's transmutation R&D programme and will support research on future innovative designs.

The operation of Superphénix confirmed the suitability of the main design and technological choices for an industrial size sodium cooled fast reactor, and substantial experience was gained in the areas of design and engineering technology. The decision to shut Superphénix down was based on the unfavourable economics of fast reactors given the slow growth in global nuclear capacity and low uranium prices during the 1990s, The permanent shutdown of Superphénix was formalized on 31 December 1998, twenty years in advance of the plant's design lifetime. Decommissioning started in 1999.

Prospective studies for future reactor systems are being carried out by the French Commissariat à l'énergie atomique (CEA) and its industrial partners. These will define the French medium and long term (beyond 2040) R&D strategy in the area of innovative nuclear systems. The strategy will have three complementary objectives: (i) the development of sodium or gas cooled fast reactors and the associated closed fuel cycle to ensure, in the long term, sustainable energy supplies through breeding and, in the medium term, to manage actinides in the spent fuel from light water reactors (LWRs); (ii) the development, in close collaboration with industrial partners, of key technologies for the nuclear production of hydrogen and the supply of high or very high temperature heat for industrial applications; (iii) further optimization of LWRs (through innovative fuels, high conversion cores, and reactor systems) to assure the most efficient use of nuclear power prior to the anticipated availability around 2040 of fourth generation fast neutron systems that are mature enough for industrial deployment. R&D priority is given to fast neutron nuclear systems with a closed fuel cycle, specifically the sodium cooled fast reactor (SFR) and the gas cooled fast reactor (GFR). These are generally recognized to be the most capable of meeting sustainability goals that include optimal use of natural uranium resources and minimized production of long lived radioactive waste.

India has limited uranium resources but about 32% of the world's thorium (Th) reserves. India has therefore developed a three-stage nuclear energy development programme, from heavy water reactors using natural uranium, through fast reactors using U and Pu, to a Th-based advanced reactor system.

The uranium available for power generation in India is about 60 000 tonnes. If this were used in pressurized heavy water reactors (PHWR), as is done presently in India, it could produce nearly 330 GW(e)-yr of electricity. This is equivalent to about 10 GW(e) of PHWRs running at a lifetime capacity factor of 80% for 40 years. The same amount of uranium, with multiple recycling using fast breeder reactors, could provide about 42 200 GW(e)-yr assuming utilization of 60% of heavy metal. This is equivalent to an installed nuclear capacity of 530 GW(e) operating for 100 years at a lifetime capacity factor of 80%. India's thorium reserves, which are greater than its uranium reserves, are estimated at about 225 000 tonnes. With multiple recycling through the appropriate reactor systems, these could produce about 150 000 GW(e), which could satisfy India's energy needs for many centuries. The expected total installed electric capacity in India in 2050 is about 1500 GW(e). Theoretically, this could be met, and fuelled from domestic resources for at least the next 100 years, through India's three-stage programme noted above: (i) PHWRs to make use of available uranium, (ii) fast reactors to convert thorium into ^{233}U, and (iii) advanced reactors (thermal and fast) to convert thorium into ^{233}U with the fissile material from steps i and ii.

Today, India is operating the Fast Breeder Test Reactor (FBTR), constructing the Prototype Fast Breeder Reactor (PFBR), and performing research and technology development for the deployment of an industrial scale fast breeder reactor (FBR).

FBTR is a 40 MW(t), 13.2 MW(e), mixed carbide fuelled, sodium cooled, loop type fast reactor with two primary and two secondary sodium loops. First criticality was achieved in October 1985. The reactor is mostly used as an irradiation facility for testing fuel and fuel elements and to gain operational experience with a liquid metal cooled fast reactor.

PFBR is a 500 MW(e), pool type, sodium cooled fast reactor with two primary pumps, four intermediate heat exchangers (IHXs) and two secondary loops. There are eight integrated steam generator (SG) units (four per secondary loop) producing steam at 766 K and 17.2 MPa. Four dedicated safety grade decay heat exchangers (SGDHR) are provided to remove the decay heat directly from the hot pool. The hot and cold pool sodium temperatures are 820 K and 670 K respectively.

PFBR is now under construction at Kalpakkam. Manufacturing of components is progressing well, and the official PFBR commissioning date is September 2010. Figure VI-2 shows the construction of the PFBR containment building as of May 2007.

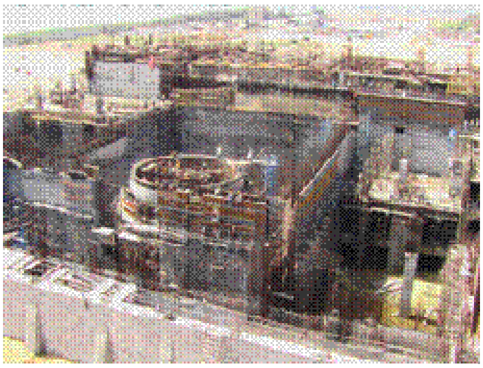

FIG. VI-2. PFBR construction. Photo courtesy of Indira Gandhi Centre for Atomic Research.

After PFBR commissioning, India's Department of Atomic Energy is planning to construct four 500 MW(e) fast breeder reactors with improved economics and enhanced safety. Features chosen to improve the economics include building pairs of reactors as twin units, reducing the main vessel diameter, incorporating an in-vessel purification system, reducing the height of the components supported on the top shield as well as of the entire reactor assembly by an improved design of the top shield, using the most cost efficient materials for construction, enhancing burnup to reduce fuel cycle costs, reducing the construction time from seven to five years, extending the designed lifetime from 40 to 60 years, and designing for a higher capacity factor. With these features, the targeted unit energy cost is 50 mils, compared to 80 mils for PFBR. As for enhanced safety features, the most important are reliable shutdown systems incorporating passive features, a passive decay heat removal system, a rationalisation of design basis events to arrive at a lower number of anticipated events, the possible elimination of core disruptive accidents, and elaborate in-service inspection and repair provisions. MOX fuel is the choice for these first four fast breeder reactors. However, for subsequent FBRs, metallic fuel would allow a higher breeding ratio and a faster expansion of nuclear power. Metallic fuel could already be introduced in one of the first four reactors once the technology, especially the fuel fabrication technology, is sufficiently mature.

Japan's Atomic Energy Commission (AEC), in October 2005, issued a *Framework for Nuclear Energy Policy* that re-iterated the significance of developing fast reactors and their associated fuel cycle technology. In March 2006, the Council for Science and Technology Policy of the Cabinet Office selected, in its third-term Science and Technology Basic Plan, fast reactors and their associated fuel cycle technology as a key technology of national importance. Subsequently, the Ministry of Education, Culture, Sports, Science and Technology (MEXT) and the Ministry of Economy, Trade and Industry (METI) published proposed action plans to develop nuclear technologies. In response to these proposals and a review by the MEXT Advisory Committee on the R&D in the Nuclear Energy Field of results of Phase-II of a *Feasibility Study on Commercialized Fast Reactor Cycle Systems*, the AEC issued in December 2006 the *Basic Policy on R&D of FBR Cycle Technologies over the Next Decade*. Its most salient points are as follows:

- MEXT, METI and the Japan Atomic Energy Agency (JAEA) will cooperate with electricity utilities, manufacturers and universities to promote R&D regarding the selected concept, and to produce, by 2015, the conceptual designs of commercial and demonstration fast breeder reactor cycle facilities that can satisfy performance criteria regarding safety, economic competitiveness, reduction of environmental burden, high efficiency in the utilization of nuclear fuel resources and enhancement of resistance to nuclear proliferation.

- JAEA shall resume operations of the prototype fast breeder power reactor MONJU in fiscal year 2008, on the precondition of safety, while promoting mutual understanding with local residents on safety issues. The goal is that, within approximately ten years of restarting MONJU, JAEA will both demonstrate MONJU's reliability as an operational fast reactor power plant and establish sodium handling technologies. After that, MONJU will be utilized for R&D activities aimed at the commercialization of fast breeder technology.

- Government and R&D organizations will also promote both the exploration and proof-of-principle activities of innovative concepts for realizing alternative fast breeder reactor cycle technologies, as well as wide-ranging relevant basic and fundamental R&D activities, utilizing their various research facilities including the experimental fast reactor JOYO.

- MEXT, METI, JAEA, utilities, and manufacturers will develop a roadmap for the commercialization of fast breeder reactor cycle technologies to enable the promotion of effective long term R&D and to facilitate the smooth transition to the demonstration phase of fast breeder reactor cycle technologies. The roadmap will specify both the requirements for the fast breeder reactor cycle technology demonstration facilities, for which the conceptual design is to be proposed in 2015, and the activities during the demonstration and commercialization phases, including a plan to achieve the construction of these demonstration facilities within the decade after 2015.

The experimental fast reactor JOYO has an MK-I breeder core and first achieved criticality in 1977. Since then, it has provided a valuable irradiation bed for advanced fuels and materials, and for improvements in fast reactor safety and operation. Future plans for using JOYO's enhanced irradiation capabilities include developing fuels and materials, enhancing safety, and irradiating fuel containing minor actinides and long-lived fission products. Since future fast reactor development work will focus on extending fuel lifetimes, JOYO will be used for tests to develop high burnup uranium-plutonium mixed oxide fuel and other innovative fuels, advanced fuel fabrication processes (e.g. vibration packing or 'VIPAC'), and long-life control rods.

MONJU, shown in Figure VI-3, is a sodium cooled, mixed uranium-plutonium oxide fuelled, loop type prototype fast power reactor rated at 280 MW(e) (714 MW(t)). MONJU's first criticality and grid connection were achieved in April 1994 and August 1995, respectively. However, in December 1995 a sodium leak occurred in the secondary heat transport system during pre-operational testing at a 40% power level. After two years of cause investigations, comprehensive safety reviews, and the necessary licensing, the permit for plant modification for countermeasures against sodium leaks was issued in December 2002 by METI. In February 2005, after receiving approval from the governor of Fukui Province, JAEA started preparatory modification work. The main modification work was approximately 94% complete by March 2007. The functional testing of the modified systems began in December 2006. Comprehensive system function tests are also planned because of the length of time the plant was shut. MONJU's restart (i.e. first criticality) is scheduled for 2008.

The conclusion of the *Feasibility Study on Commercialized Fast Reactor Cycle Systems* that began in July 1999 was that the most viable technology options for commercialization were: the plutonium-uranium mixed oxide fuelled, sodium cooled fast breeder reactor, with fuel fabrication based on simplified pelletizing, and with advanced aqueous reprocessing. The runner-up concept, deemed equally viable for commercialization but more uncertain in its social and technical aspects, was a metallic plutonium-uranium fuelled, sodium cooled fast breeder reactor, with fuel fabrication based on injection casting, and with electro-refining reprocessing. In 2006, JAEA launched the Fast Reactor Cycle Technology Development (FaCT) project focused on R&D for the first, most viable concept.

FIG. VI-3. MONJU. Photo courtesy of JAEA.

In the **Republic of Korea**, fast reactor technology development dates back to 1992 when the Korea Atomic Energy Commission approved a long-term R&D plan for a sodium cooled fast reactor. In February 2007, the development of the conceptual design of KALIMER-600, an advanced sodium cooled fast reactor concept was completed by the Korea Atomic Energy Research Institute (KAERI). The core is loaded with single enrichment metal fuels and configured without blanket assemblies. To achieve power flattening in the single-enrichment core the current design contains different cladding thicknesses in the inner, middle and outer core regions. KALIMER-600 is a pool type reactor, which has a large heat capacity and can reduce any rapid transients caused by reactor trips. The heat transport system consists of a primary heat transport system (PHTS), an intermediate heat transport system (IHTS), a residual heat removal system (RHRS), and a steam generating system (SGS). The PHTS consists of the PHTS pump, intermediate heat exchangers (IHXs) and all the internal structures within the reactor and the containment vessel. The PHTS pump is a centrifugal mechanical pump. The IHTS consists of piping and an electro-magnetic pump. The RHRS has three layers and consists of the passive decay-heat removal circuit (PDRC), the intermediate reactor auxiliary cooling system (IRACS), and the steam/feed water system. When normal decay heat removal is not available through the steam/feed water system, the operator can activate the IRACS. The IRACS is operated by closing the SG isolation valve and by opening the IRACS isolation valve. Then the air heat exchanger (AHX) of the IRACS dumps heat to the atmosphere. In the event of a station blackout, the KALIMER-600 design relies on the PDRC. The PDRC is a pure passive system relying exclusively on natural convection phenomena, without any operator action and active components.

The main features of the mechanical design in KALIMER-600 are the seismically isolated reactor building, the reduced total pipe length of the IHTS, the simplified reactor support, and the compact reactor internal structures.

In the **Russian Federation** there are two fast reactors in operation, the experimental reactor BOR-60 at Dimitrovgrad, and BN-600, the commercial Unit 3 of the Beloyarsk nuclear power plant.

BOR-60 has been in operation for more than 36 years. While also producing heat and electricity, BOR-60 is used for material tests, isotope production, and fast reactor equipment tests. The reactor has an operational licence until 31 December 2009, and lifetime extension activities are currently underway.

BN-600 (Figure VI-4) has been in operation for more than 27 years. BN-600 is the largest operating fast reactor power unit in the world. It is a sodium cooled loop type reactor using mixed oxide fuel, and its safety performance and operating reliability have been excellent. As of 31 December 2006, BN-600 had produced more than 100.4 TWh of electricity, of which 4.13 TWh were generated in 2006. In 2006, BN-600's load factor was 78.6%, while the average load factor since 1983 is 74.2%. The design lifetime of the BN-600 reactor plant expires in April 2010, and lifetime extension activities are also underway for this reactor.

FIG. VI-4. BN-600 at the Beloyarsk NPP site. Photo courtesy of the Institute of Physics and Power Engineering.

BN-800 is under construction also at the Beloyarsk site. BN-800 is a sodium cooled, uranium-plutonium mixed oxide fuelled fast reactor. Although originally designed for 800 MW(e), BN-800's design power has been increased to 880 MW(e) while the thermal power remains unchanged (at 2100 MW) through optimizing the third circuit, improving the turbine parameters, and eliminating steam extraction for heating. These and other design improvements have also extended the reactor's lifetime from 30 to 40 years. The pouring of concrete for the foundations of the reactor compartment and turbine hall has been completed. Manufacturing of large components (e.g. the main and safety reactor vessels, and primary circuit sodium tanks) has started. Commissioning of BN-800 is foreseen for 2012. Vibro-packed fuel is being considered on the basis of successful testing in BN-600.

The Russian Federation's advanced fast reactor research and technology development programmes are currently focused on the development of advanced sodium cooled fast reactors and of fast reactors cooled by heavy liquid metal coolants (i.e. lead and lead-bismuth).

The development of advanced sodium cooled fast reactors involves two main activities: R&D work to develop the conceptual design of a large commercial fast sodium cooled reactor (BN-1800), and the conceptual development of a small, modular and transportable two-circuit sodium cooled fast reactor with a gas turbine (BN GT nuclear co-generation power plant (NCPP)).

In the area of heavy liquid metal cooled fast reactors, work is concentrated on R&D for the lead cooled BREST OD 300 reactor, and on development work for the basic design of the lead-bismuth cooled 'SVBR-75/100 reactor facility (RF)'. The BREST OD 300 design substantiation work covers coolant technology, material studies and safety issues.

In the **United States of America**, the Department of Energy's NP2010 programme facilitates the deployment of advanced light water reactors (ALWRs) while its Global Nuclear Energy Partnership (GNEP) seeks to develop, demonstrate, and deploy advanced technology for recycling spent nuclear fuel that provides important waste management and proliferation benefits, including no separation of plutonium. The development of fast reactor technology is a major part of this strategy. Using sodium cooled fast reactors with a closed fuel cycle could also result in better utilization of the US geologic repository. The GNEP fuel cycle strategy is based on expanded nuclear power production in ALWRs with the spent fuel separated into several components for tailored waste management. The trans-uranium nuclides would be recycled to be consumed in advanced reactors for further power production. Fast reactors would be utilized for closed recycling and 'burning' of these materials. The development of recycled fuels and the management of the waste from ALWRs and fast reactor recycling are additional important technological challenges. The overall strategy focuses on waste management and non-proliferation benefits.

C.2. International fast reactor programmes

The challenges of developing fast reactors and the closed fuel cycle exceed in many cases the capabilities of individual countries. To pool international experience, expertise and resources, several international initiatives have been established to jointly evaluate and develop promising options.

The Generation IV International Forum (GIF) is a joint initiative of thirteen countries. Six innovative reactors have been selected for further development and potential deployment by about 2030. Three of these design concepts are fast reactors cooled by sodium, lead (or lead-bismuth) or helium. A fourth, the super-critical water reactor, includes an option to use fast neutrons. GIF was created in May 2001 to lead the collaborative efforts to develop the next generation of nuclear power plants. It first established development goals (specifically sustainability, economics, safety and reliability, proliferation-resistance and physical protection) and a roadmap for deployment around 2030. Current development work is focused on materials, systems and components.

The IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) is a discussion forum for experts and policy makers on all aspects of nuclear energy planning as well as on the development and deployment of innovative nuclear energy systems (INSs). It currently has 28 Members and brings together technology holders, users and potential users to consider jointly the international and national actions required for achieving desired innovations in nuclear reactors and fuel cycles. INPRO pays particular attention to the needs of developing countries. INPRO has developed a methodology for assessing INSs and completed assessment studies in which the methodology was applied by interested Member States. For example, a joint assessment study of a closed nuclear fuel cycle with fast reactors (CNFC-FR) was done by Canada, China, France, India, Japan, Republic of Korea, Russian Federation, and Ukraine in 2005-2007. The study assessed the CNFC-FR against sustainability criteria to determine milestones for deployment and to establish a framework for collaborative R&D. The study helped identify elements of the fast reactor technology development strategy that would meet sustainable development requirements of nuclear power. Currently, the IAEA coordinates and supports collaborative projects identified by INPRO Members to address technology improvement in the areas of safety, economics, environmental impacts, waste, proliferation resistance and infrastructure.

The Global Nuclear Energy Partnership (GNEP) is an initiative by the USA, which, as part of its efforts described earlier, seeks to develop cooperative efforts to expand the use of nuclear energy. GNEP had 19 members as of the end of 2007, and its current approach is to develop industry-led prototype facilities for fast reactor and LWR spent fuel separation relatively quickly, i.e. by about 2020. These facilities would demonstrate the innovative technologies and design features important for subsequent commercial demonstration plants and would substantiate fuel cycle benefits. Concurrently, a DOE laboratory-led technology development and research programme will continue. This will focus on innovative fuel cycle technologies including fuel development for recycling. For fast reactor research, the critical focus will be on capital cost improvements, principally through design simplifications, new technology, and advanced simulation.

D. Conclusions

Many scenarios of possible energy futures foresee an important role for nuclear power. While some explore the impacts of a nuclear phase out, others envision a major growth in nuclear power's share of the world energy mix. A growing world population, continued economic development, concerns about greenhouse gas emissions, rising fossil fuel prices and energy supply security are the main drivers of current rising expectations for nuclear power. To make nuclear power a truly long term option, fast reactors are needed to produce new fissile material and to reduce the amount of waste and its impact on the environment. The fast reactor concept is not new; several reactors have operated in various countries. Globally fast reactors did not break through commercially because demand for new fissile material was lower than had been anticipated, and the fast reactor was not economic for electricity production. However for some countries, notably India, the restricted availability of uranium has made fast reactors more attractive in the short term.

Currently, several fast reactor concepts are being developed with various coolants and fuel cycles. Major challenges are still to reduce costs and to develop materials and fuels for commercial operation. Different types of reactors and fuel cycles may be appropriate in different countries.

The various fast reactor research and technology development programmes are not uniform because they reflect different needs. The renewed interest in fast reactors, as indicated by increased funding and the many national and international fast reactor research and technology development activities, is driven to different degrees by resource scarcity, security of supply considerations, and waste management concerns.