World Nuclear Association

Fast Neutron Reactors

(*updated April 2010*)

- **Fast neutron reactors are a technological step beyond conventional power reactors.**
- **They offer the prospect of vastly more efficient use of uranium resources and the ability to burn actinides which are otherwise the long-lived component of high-level nuclear wastes.**
- **Some 390 reactor-years experience has been gained in operating them**

About 20 Fast Neutron Reactors (FNR) have already been operating, some since the 1950s, and some supplying electricity commercially. About 390 reactor-years of operating experience have been accumulated. Fast reactors more deliberately use the uranium-238 as well as the fissile U-235 isotope used in most reactors. If they are designed to produce more plutonium than they consume, they are called Fast Breeder Reactors (FBR). But many designs are net consumers of fissile material including plutonium.* Fast neutron reactors also can burn long-lived actinides which are recovered from used fuel out of ordinary reactors.

* If the ratio of final to initial fissile content is less than 1 they are burners, consuming more fissile material (U-235, Pu and minor actinides) than they produce (fissile Pu), if more than 1 they are breeders. This is the burn ratio or breeding ratio.

Several countries have research and development programs for improved Fast Neutron Reactors, and the IAEA's INPRO program involving 22 countries (see later section) has fast neutron reactors as a major emphasis, in connection with closed fuel cycle. For instance one scenario in France is for half of the present nuclear capacity to be replaced by fast neutron reactors by 2050 (the first half being replaced by 3rd-generation EPR units).

The FNR was originally conceived to burn uranium more efficiently and thus extend the world's uranium resources - it could do this by a factor of about 60. When those resources were perceived to be scarce, several countries embarked upon extensive FBR development programs. However significant technical and materials problems were encountered, and also geological exploration showed by the 1970s that uranium scarcity would not be a concern for some time. Due to both factors, by the 1980s it was clear that FNRs would not be commercially competitive with existing light water reactors for some time.

Today there has been progress on the technical front, but the economics of FNRs still depends on the value of the plutonium fuel which is bred and used, relative to the cost of fresh uranium. Also there is international concern over the disposal of ex-military plutonium, and there are proposals to use fast reactors (as "burners") for this purpose. In both respects the technology is important to long-term considerations of world energy sustainability.

Fast Neutron Reactors

Output:	MWe	MW (thermal)	Operation
USA			
EBR 1	0.2		1951-63
EBR II	20	62.5	1963-94
Fermi 1	66		1963-72
SEFOR		20	1969-72
Fast Flux Test Facility		400	1980-93

UK			
Dounreay FR	15		1959-77
Protoype FR	270		1974-94
France			
Rapsodie		40	1966-82
Phenix*	250		1973-2009
Superphenix 1	1240		1985-98
Germany			
KNK 2	21		1977-91
India			
FBTR		40	1985-
Japan			
Joyo		140	1978-
Monju	280		1994-96-?
Kazakhstan			
BN 350*	135		1972-99
Russia			
BR 5 /10 Obninsk		5 /8	1959-71, 1973-
BOR 60 Dimitrovgrad	12		1969-
BN 600* Beloyarsk	560		1980-

FNR operation

Natural uranium contains about 0.7% U-235 and 99.3% U-238. In any reactor some of the U-238 component is turned into several isotopes of plutonium during its operation. Two of these, Pu-239 and Pu-241, then undergo fission in the same way as U-235 to produce heat. In a FNR this process is optimised so that it 'breeds' fuel. Hence FNRs can utilise uranium about 60 times more efficiently than a normal reactor. They are however expensive to build and operate, including the reprocessing, and are only justified economically if uranium prices remain above 1990s low levels.

see also American Nuclear Society position statement, November 2005 (pdf).

The fast reactor has no moderator and relies on fast neutrons alone to cause fission, which for uranium is less efficient than using slow neutrons. Hence a fast reactor usually uses plutonium as its basic fuel, since it fissions sufficiently with fast neutrons to keep going*. At the same time the number of neutrons produced per fission is 25% more than from uranium, and this means that there are enough (after losses) not only to maintain the chain reaction but also continually to convert U-238 into more Pu-239. Furthermore, the fast neutrons are more efficient than slow ones in doing this breeding. These are the main reasons for avoiding the use of a moderator. The coolant is a liquid metal (normally sodium) to avoid any neutron moderation and provide a very efficient heat transfer medium. So, the fast reactor "burns" and "breeds" fissile plutonium.**

* high-enriched uranium (over 20% U-235) would fission, too. At this concentration of U-235, the cross-section for fission with fast neutrons is sufficient to sustain the chain-reaction despite less likelihood of fission, so about 20% of fissile nuclei is required in the fuel. Up to 20% U is actually defined as "low-enriched' uranium.
** Both U-238 and Pu-240 are "fertile" (materials), i.e. by capturing a neutron they become (directly or indirectly) fissile Pu-239 and Pu-241 respectively.

The conventional fast reactors built so far are generally fast breeder reactors (FBRs) implying a net increase in Pu-239 from breeding. These have a "fertile blanket" of depleted uranium (U-238) around the core, and this is where much of the Pu-239 is produced. Neutron activity is very low in the blanket, so the plutonium produced there remains almost pure Pu-239 - largely not burned or changed to Pu-240. The blanket can then be reprocessed (as is the core) and the plutonium recovered for use in the core, or for further FNRs.* However, apart from India, there are apparently no plans to build any more fast reactors with this design; fast reactor concepts being developed for the Generation IV program will simply have a core so that the plutonium production and consumption both occur there. Russia's BREST is the most advanced design. Conceptually, refuelling means simply adding a little natural or depleted uranium – about one or two percent of the total required for a comparable light water reactor. Due to the high radiation levels in the core, using simply a core and no blanket gives rise to some new challenges in how the fuel is fabricated and managed.

* Operation of the BN-600 reactor to burn weapons-grade plutonium from 2012, will have the breeding blanket of depleted uranium removed and replaced by stainless steel reflector assemblies.

Many core configurations are possible, but for maximum breeding, the conventional core plus blanket arrangement is best. If a breeding ratio of less than 1, or just a little more than 1 is wanted, then axial blankets which are included in the fuel pins will serve the purpose. The entire fuel pins are then reprocessed, and the newly-formed plutonium is mixed with the used fuel materials from the fissile zone of the pins. It is also possible to have a uniform core without separate U-238, and with stainless steel reflectors, but little breeding is then possible.

India's three-stage thorium fuel cycle is unique, and still under development (see next section). Here, fast breeder reactors form stage 2 and use plutonium-based fuel in the core to breed both U-233 from thorium and Pu-239 from U-238 in the blanket. The plutonium and U-233 is needed as a driver fuel in advanced heavy water reactors forming stage 3 of the concept – these get about 75% of their power from the thorium, but need the plutonium and U-233 to do so.

The core of a fast reactor is much smaller than that of a normal nuclear reactor, and it has a higher power density, requiring very efficient heat transfer. For instance, the core of Russia's BN-600 reactor (560 MWe) is 2 metres high and 0.75 m diameter. Fuel may be enriched uranium oxide (BN-350, BN-600) or MOX (BOR-60, BN-800). BREST will use a high-density U+Pu nitride fuel with no requirement for high enrichment levels.

One effect of the 1980s halt to FNR development is that separated plutonium (from reprocessing used light water reactor fuel) which was originally envisaged for FNRs is now being used as mixed oxide (MOX) fuel in conventional reactors.

Fast neutron reactors have a high power density and are normally cooled by liquid metal such as sodium, lead, or lead-bismuth, with high conductivity and boiling point and no moderating effect. They operate at around 500-550°C at or near atmospheric pressure. Fast reactors typically use boron carbide control rods.

In some respects a liquid metal coolant is more benign overall than very high pressure water, which requires robust engineering on account of the pressure. However, the design needs to ensure that there is no chemical interaction (eg sodium-water), and is lead-cooled, the materials used need to allow for molten lead being very corrosive. Some future plans are for gas-cooled fast reactors.

Also fast reactors have a strong negative temperature coefficient (the reaction slows as the temperature rises unduly), an inherent safety feature, and the basis of automatic load following in many new designs.

Experiments on a 19-year old UK breeder reactor before it was decommissioned in 1977, and on EBR-2 in the USA in 1986, showed that the metal fuel with liquid sodium cooling system made them less sensitive to coolant failures than the more conventional very high pressure water and steam systems in light water reactors. More recent operating experience with large French and UK prototypes has confirmed this. With loss of coolant flow they simply shut themselves down.

There is renewed interest in fast reactors due to their ability to fission actinides, including those which may be recovered from ordinary reactor used fuel. The fast neutron environment minimises neutron capture reactions and maximises fissions in actinides. This means less long-lived nuclides in high-level wastes (the fission products being preferable due to shorter lives).

Fast reactor fuel cycles

Reprocessing used fuel, and especially the blanket assemblies, is fundamental to the FBR fuel cycle. Typically the recovered plutonium from aqueous reprocessing is incorporated into the core as MOX fuel and any surplus deployed elsewhere. The general principles of this are described above.

However, with the transition from core and blanket designs to integrated core designs, it is likely that used fuel will be reprocessed using electrometallurgical processes (so-called pyro-processing) and plutonium will not be separated but will remain with some highly radioactive isotopes.

See also: Processing used nuclear fuel for recycle paper.

India's nuclear power program has been focused on developing an advanced heavy-water thorium cycle, based on converting abundant thorium-232 into fissile uranium 233. The first stage of this employs PHWRs fuelled by natural uranium, and light water reactors, to produce plutonium. Stage two uses fast neutron reactors burning the plutonium to breed U-233 from thorium. The blanket around the core will have uranium as well as thorium, so that further plutonium (ideally high-fissile Pu) is produced as well as the U-233. Then in stage three, advanced heavy water reactors burning the U-233 and this plutonium as driver fuels, but utilising thorium as their main fuel, and getting about two thirds of their power from the thorium.

A 500 MWe prototype fast breeder reactor (PFBR) is under construction at Kalpakkam and is expected to be operating in 2011, fuelled with uranium-plutonium oxide or carbide. It will have a blanket with thorium and uranium to breed fissile U-233 and plutonium respectively. Initial FBRs will have mixed oxide fuel but these will be followed by metallic fuelled ones to enable shorter doubling time.

A reprocessing centre for thorium fuels is being set up at Kalpakkam.

Europe

France operated its **Phenix** fast reactor from 1973 to 2009, apart from a few years for refurbishing. It ceased generating power early in 2009 but ran until October 2009 as a research reactor. Closure of the 1250 MWe commercial prototype **Superphenix** FBR in 1998 on political grounds after very little operation over 13 years set back developments. Research work on the 1450 MWe European FBR has almost ceased.

In mid 2006 the French Atomic Energy Commission (CEA) was commissioned by the government to development two types of fast neutron reactors which are essentially Generation IV designs: an improved version of the sodium-cooled type which already has 45 reactor-years operational experience in France, and an innovative gas-cooled type. Both would have fuel recycling, and in mid 2009 it was recommended that the sodium-cooled model, **Astrid** (Advanced Sodium

Technological Reactor for Industrial Demonstration), should be a high priority in R&D on account of its actinide-burning potential. The CEA is seeking support under the EC's European Sustainable Nuclear Industrial Initiative and partnerships with Japan and China to develop the sodium-cooled model. However, it notes that China (like India) is aiming for high breeding ratios to produce enough plutonium to crank up a major push into fast reactors.

Astrid is envisaged as a 600 MWe prototype of a commercial series which is likely to be deployed from about 2050. It will have high fuel burnup, including minor actinides in the fuel elements, and use an intermediate sodium loop, though whether the tertiary coolant is water/steam or gas is an open question. Four independent heat exchanger loops are likely, and it will be designed to reduce the probability and consequences of severe accidents to an extent that is not now done with FNRs.

Astrid is called a "self-generating" fast reactor rather than a breeder in order to demonstrate low net plutonium production. Astrid is designed to meet the stringent criteria of the Generation IV International Forum in terms of safety, economy and proliferation resistance. CEA plans to build it at Marcoule.

The Astrid program includes development of the reactor itself and associated fuel cycle facilities: a dedicated MOX fuel fabrication line (possibly in Japan) and a pilot reprocessing plant for used Astrid fuel. The program also includes a workshop for fabricating fuel rods containing actinides for transmutation, called Alfa, scheduled to operate in 2023, though fuel containing minor actinides would not be loaded for transmutation in Astrid before 2025.

The second line of French FNR development is the gas-cooled fast reactor. A 50-80 MWt experimental version - **Allegro** - is envisaged by 2020. This will have either a ceramic core with 850°C outlet temperature, or a MOX core at 560°C. The secondary circuit will be pressurized water.

In the UK, the **Dounreay** Fast Reactor started operating in 1959 using sodium-potassium coolant. This was followed there by the much larger Prototype Fast Reactor which operated for 20 years until the government withdrew funding.

Russia, Kazakhstan

The Russian **BN-600** fast breeder reactor - Beloyarsk unit 3 - has been supplying electricity to the grid since 1980 and is said to have the best operating and production record of all Russia's nuclear power units. It uses chiefly uranium oxide fuel, some enriched to over 20%, with some MOX in recent years. The sodium coolant delivers 550°C at little more than atmospheric pressure. Russia plans to reconfigure the BN-600 by replacing the fertile blanket around the core with steel reflector assemblies to burn the plutonium from its military stockpiles and to extend its life beyond the 30-year design span.

The **BN-350** prototype FBR generated power in Kazakhstan for 27 years to 1999 and about half of its 1000 MW(thermal) output was used for water desalination. It used uranium enriched to 17-26%. Its design life was 20 years, and after 1993 it operated on the basis of annual licence renewal. Russia's **BOR-60** was a demonstration model preceding it.

Construction is well advanced on Beloyarsk-4 which is the first **BN-800** from OKBM Afrikantov, a new, more powerful (880 MWe) FBR, which is actually the same overall size as BN-600.   It has improved features including fuel flexibility - U+Pu nitride, MOX, or metal, and with breeding ratio up to 1.3. However, during the plutonium disposition campaign it will be operated with a breeding ratio of less than one. It has much enhanced safety and improved economy - operating cost is expected to be only 15% more than VVER. It is capable of burning up to 2 tonnes of plutonium per year from dismantled weapons and will test the recycling of minor actinides in the fuel.

In 2009 two BN-800 reactors were sold to China, with construction due to start in 2011. The precise core design of these is not known.

Russia has experimented with several lead-cooled reactor designs, and has used lead-bismuth cooling for 40 years in reactors for its Alfa class submarines. Pb-208 (54% of naturally-occurring lead) is transparent to neutrons. A significant new Russian design from NIKIET is the **BREST** fast neutron reactor, of 300 MWe or more with lead as the primary coolant, at 540°C, and supercritical steam generators. It is inherently safe and uses a U+Pu nitride fuel. No weapons-grade plutonium can be produced (since there is no uranium blanket), and spent fuel can be recycled indefinitely, with on-site facilities. A pilot unit is planned at Beloyarsk and 1200 MWe units are proposed.

A smaller and newer Russian design is the Lead-Bismuth Fast Reactor (**SVBR**) of 75-100 MWe. This is an integral design, with the steam generators sitting in the same Pb-Bi pool at 400-495°C as the reactor core, which could use a wide variety of fuels. The unit would be factory-made and shipped as a 4.5m diameter, 7.5m high module, then installed in a tank of water which gives passive heat removal and shielding. A power station with 16 such modules is expected to supply electricity at lower cost than any other new Russian technology as well as achieving inherent safety and high proliferation resistance. (Russia built 7 Alfa-class submarines, each powered by a compact 155 MWt Pb-Bi cooled reactor, and 70 reactor-years operational experience was acquired with these.) In 2008 Rosatom and the Russian Machines Co put together a joint venture to build a prototype 100 MWe SVBR reactor.

Rosatom has put forward two fast reactor implementation options for government decision in relation to the Advanced Nuclear Technologies Federal Program 2010-2020. The first focuses on a lead-cooled fast reactor such as BREST with its fuel cycle, and assumes concentration of all resources on this project with a total funding of about RUR 140 billion (about $3.1 billion). The second scenario assumes parallel development of fast reactors with lead, sodium and lead-bismuth coolants and their associated fuel cycles. It would cost about RUR 165 billion ($4.7 billion). The second scenario is viewed as the most favoured, since it is believed to involve lower risks than the first one. If implemented it would result in technical designs of the Generation IV reactor and associated closed fuel cycles technologies by 2013, and a technological basis of the future innovative nuclear energy system featuring the Generation IV reactors working in closed fuel cycles by 2020.

Japan

A significant part of Japanese energy policy has been to develop FBRs in order to improve uranium utilisation dramatically. From 1961 to 1994 there was a strong commitment to FBRs, but in 1994 the FBR commercial timeline was pushed out to 2030, and in 2005 commercial FBRs were envisaged by 2050.

In 1999 Japan Nuclear Cycle Development Institute (JNC) initiated a program to review promising concepts, define a development plan by 2005 and establish a system of FBR technology by 2015. The parameters are: passive safety, economic competitiveness with LWR, efficient utilisation of resources (burning transuranics and depleted U), reduced wastes, proliferation resistance and versatility (include hydrogen production). Utilities are also involved.

Phase 2 of the study focused on four basic reactor designs: sodium-cooled with MOX and metal fuels, helium-cooled with nitride and MOX fuels, lead-bismuth eutectic-cooled with nitride and metal fuels, and supercritical water-cooled with MOX fuel. All involve closed fuel cycle, and three reprocessing routes were considered: advanced aqueous, oxide electrowinning and metal pyroprocessing (electrorefining). This work is linked with the Generation IV initiative, where Japan is playing a leading role with sodium-cooled FBRs.

Japan's **Joyo** experimental reactor which has been operating since 1977 is now being boosted to 140 MWt.

The 280 MWe **Monju** prototype FBR reactor started up in April 1994, but a sodium leakage in its secondary heat transfer system during performance tests in 1995 meant that it was shut down and has not operated since. It produced 246 MWe when it was operating. Its oversight has passed to JNC, and the Minister for Science & Technology has said that its early restart is a key aim. A Supreme court decision in May 2005 cleared the way for restarting it in 2008, but this has been put back to 2010.

Mitsubishi Heavy Industries (MHI) is involved with a consortium to build the **Japan Standard Fast Reactor (**JSFR) concept, with breeding ratio less than 1. This is a large unit which will burn actinides with uranium and plutonium in oxide fuel. It could be of any size from 500 to 1500 MWe. In this connection MHI has also set up Mitsubishi FBR Systems (MFBR).

Japan's **LSPR** is a lead-bismuth cooled reactor design of 150 MWt /53 MWe. Fuelled units would be supplied from a factory and operate for 30 years, then be returned. Concept intended for developing countries.

A small-scale design developed by Toshiba Corporation in cooperation with Japan's Central Research Institute of Electric Power Industry (CRIEPI) and funded by the Japan Atomic Energy Research Institute (JAERI) is the 5 MWt, 200 kWe **Rapid-L**, using lithium-6 (a liquid neutron poison) as control medium. It would have 2700 fuel pins of 40-50% enriched uranium nitride with 2600°C melting point integrated into a disposable cartridge. The reactivity control system is passive, using lithium expansion modules (LEM) which give burnup compensation, partial load operation as well as negative reactivity feedback. As the reactor temperature rises, the lithium expands into the core, displacing an inert gas. Other kinds of lithium modules, also integrated into the fuel cartridge, shut down and start up the reactor. Cooling is by molten sodium, and with the LEM control system, reactor power is proportional to primary coolant flow rate. Refuelling would be every 10 years in an inert gas environment. Operation would require no skill, due to the inherent safety design features. The whole plant would be about 6.5 metres high and 2 metres diameter.

The Super-Safe, Small & Simple - **4S 'nuclear battery'** system is being developed by Toshiba and CRIEPI in Japan in collaboration with STAR work in USA. It uses sodium as coolant (with electromagnetic pumps) and has passive safety features, notably negative temperature and void reactivity. The whole unit would be factory-built, transported to site, installed below ground level, and would drive a steam cycle. It is capable of three decades of continuous operation without refuelling. Metallic fuel (169 pins 10mm diameter) is uranium-zirconium or U-Pu-Zr alloy enriched to less than 20%. Steady power output over the core lifetime is achieved by progressively moving upwards an annular reflector around the slender core (0.68m diameter, 2m high). After 14 years a neutron absorber at the centre of the core is removed and the reflector repeats its slow movement up the core for 16 more years. In the event of power loss the reflector falls to the bottom of the reactor vessel, slowing the reaction, and external air circulation gives decay heat removal.

Both 10 MWe and 50 MWe versions of 4S are designed to automatically maintain an outlet coolant temperature of 510°C - suitable for power generation with high temperature electrolytic hydrogen production. Plant cost is projected at US$ 2500/kW and power cost 5-7 cents/kWh for the small unit - very competitive with diesel in many locations. The design has gained considerable support in Alaska and and toward the end of 2004 the town of Galena granted initial approval for Toshiba to build a 4S reactor in that remote location. A pre-application NRC review is under way with a view to application for design certification in October 2010 (delayed from 2009 by NRC workload), and construction and operating licence (COL) application to follow. Its design is sufficiently similar to PRISM - GE's modular 150 MWe liquid metal-cooled inherently-safe reactor which went part-way through US NRC approval process for it to have good prospects of licensing.

The L-4S is Pb-Bi cooled version of 4S.

India

In India, research continues. At the Indira Gandhi Centre for Atomic Research a 40 MWt fast breeder test reactor (**FBTR**) has been operating since 1985. In addition, the tiny **Kamini** there is employed to explore the use of thorium as nuclear fuel, by breeding fissile U-233.

In 2002 the regulatory authority issued approval to start construction of a 500 MWe prototype fast breeder reactor (PFBR) at **Kalpakkam** and this is now under construction by BHAVINI. It is expected to be operating in 2011, fuelled with uranium-plutonium oxide or carbide (the reactor-grade Pu being from its existing PHWRs) and with a thorium blanket to breed fissile U-233. This will take India's ambitious thorium program to stage 2, and set the scene for eventual full utilisation of the country's abundant thorium to fuel reactors. Four more such fast reactors have been announced for construction by 2020. Initial Indian FBRs will be have mixed oxide fuel but these will be followed by metallic-fuelled ones to enable shorter doubling time.

Indian figures for PHWR reactors using unenriched uranium suggest 0.3% utilization, which is contrasted with 75% utilization expected for PFBR.

China

In China, R&D on fast neutron reactors started in 1964. A 65 MWt fast neutron reactor - the Chinese Experimental Fast Reactor (**CEFR**) - is under construction near Beijing by Russia's OKBM Afrikantov in collaboration with OKB Gidropress, NIKIET and Kurchatov Institute. It is reported to have a 25 MWe turbine generator and is expected to achieve first criticality in April 2010. A 600 MWe prototype fast reactor was envisaged by 2020 and there was talk of a 1500 MWe one by 2030. CNNC expects FNR technology to become predominant by mid century.

However, in October 2009 an agreement was signed with Russia's Atomstroyexport to start pre-project and design works for a commercial nuclear power plant with two **BN-800** reactors in China, with construction to start in August 2011. These would be similar to the OKBM Afrikantov design being built at Beloyarsk 4 and due to start up in 2012. In June 2009 St Petersburg Atomenergopoekt said it was starting design work on a BN-800 reactor for China, with two proposed at coastal sites. The project is expected to lead to bilateral cooperation of fuel cycles for fast reactors.

USA

In the USA, five fast neutron reactors have operated, and several more designed. The experimental breeder reactor **EBR-1** at Idaho in 1951 produced enough power to run its own building - a milestone achievement.

The **EBR-II** was a demonstration reactor - 62.5 MW thermal, and it typically operated at 19 MWe, providing heat and power to the Idaho facility over 1963-94. The idea was to demonstrate a complete sodium-cooled breeder reactor power plant with on-site reprocessing of metallic fuel, and this was successfully done 1964-69. The emphasis then shifted to testing materials and fuels (metal and ceramic oxides, carbides and nitrides of U & Pu) for larger fast reactors. Finally it became the IFR prototype, using metallic alloy U-Pu-Zr fuels. All the time, it generated some 1 TWh of power as well.

The EBR-II was the basis of the US **Integral Fast Reactor (IFR)** program, considered by the National Academy of Sciences to be the nation's highest priority research for future reactor types. This was developing a fully-integrated system with electrometallurgical 'pyroprocessing', fuel fabrication and fast reactor in same complex*. The reactor could be operated as a breeder or not. Some $46 million of the IFR funding was provided by a Japanese utility consortium.

* So far the only electrometallurgical technique that has been licensed for use on a significant scale is the IFR electrolytic process developed by Argonne National Laboratory and used for pyroprocessing the used fuel from the EBR-II experimental fast reactor which ran from 1963-1994. This is essentially a partitioning-conditioning process, because neither plutonium nor other transuranics are recovered for recycle. The process is used to facilitate the disposal of a fuel that could not otherwise be sent directly to a geologic repository.

IFR program goals were demonstrating inherent safety apart from engineered controls,* improved management of high-level nuclear wastes by recycling all actinides, so that only fission products remain as HLW,** and using the full energy potential of uranium rather than only about one percent of it. All these were demonstrated, though the program was aborted before the recycle of neptunium and americium was properly evaluated. IFR fuel first used in 1986 reached 19% burnup (compared with 3-4% for conventional reactors), and 22% was targeted.

* In April 1986, two tests were performed on the EBR-II. In the first, the main primary cooling pumps were shut off with the reactor at full power. Without allowing the normal shutdown systems to interfere, the reactor power dropped to near zero within about five minutes. No damage to the fuel or the reactor resulted. The second test was again with the reactor at full power, and the flow in the secondary cooling system was stopped. This caused the reactor temperature to increase, and as the fuel, primary sodium coolant and structure expanded, the reactor shut down on its own.

** for a 1000 MWe plant at 90% capacity factor about 990kg/yr of HLW is projected.

A further political goal was demonstrating a proliferation-resistant closed fuel cycle, with plutonium being recycled with other actinides.

In 1994, Congress under the Clinton administration shut EBR-II down, delivering a major setback to FNR fuel cycle developments. The IFR program is now being reinvented as part of the Advanced Fuel Cycle Initiative (see below), while EBR-II is being decommissioned. An EBR-III of 200-300 MWe was proposed but not developed.

The first US commercial FBR was **Fermi-1** in Michigan, but it operated for only three years before a coolant problem caused overheating and it was shut down with some damage to the fuel. After repair it was restarted in 1970, but its licence was not renewed in 1972.

The Southeast Experimental Fast Oxide Reactor (**SEFOR**) was built in 1965 and operated for three years in Arkansas by GE under contract to the US government. It was the only fast reactor to use a full core of Pu-U mixed oxide fuel, and was sodium-cooled. It completed its safety test program in 1972, demonstrating the capability of the Doppler coefficient (re core thermal expansion) in a mixed oxide reactor to stabilise it and control accidents in oxide-fueled, sodium-cooled fast reactors. Fuel and coolant were removed in 1972 and the University of Arkansas bought it in 1975.

The 400 MWt **Fast Flux Test Facility** was in full operation 1982-92 at Hanford as a major national research reactor. It was closed down at the end of 1993, and since 2001 it has been deactivated under care and maintenance pending possible decommissioning. However, in August 2006 the Department of Energy indicated that it could possibly be recommissioned as part of the Global Nuclear Energy Partnership demonstration process.

GE with the DOE national laboratories has been developing a modular liquid metal-cooled inherently-safe reactor - PRISM during the advanced liquid-metal fast breeder reactor (ALMR) program. No US fast neutron reactor has so far been larger than 66 MWe and none has supplied electricity commercially.

Today's **PRISM** is a GE-Hitachi design for compact modular pool-type reactors with passive cooling for decay heat removal. After 30 years of development it represents GEH's Generation IV solution to closing the fuel cycle in the USA. Each PRISM Power Block consists of two modules of 311 MWe each, operating at high temperature – over 500°C. The pool-type modules below ground level contain the complete primary system with sodium coolant. The Pu & DU fuel is metal, and obtained from used light water reactor fuel. However, all transuranic elements are removed together in the electrometallurgical reprocessing so that fresh fuel has minor actinides with the plutonium. Fuel stays in the reactor about six years, with one third removed every two years, and breeding ratio is 0.8. Used PRISM fuel is recycled after removal of fission products. The commercial-scale plant concept, part of a Advanced Recycling Centre, uses three power blocks (six reactor modules) to provide 1866 MWe. See also electrometallurgical section in Processing Used Nuclear Fuel paper.

The **Encapsulated Nuclear Heat Source** (ENHS) concept is a liquid metal-cooled reactor of 50 MWe being developed by the University of California. The core is in a metal-filled module sitting in a large pool of secondary molten metal coolant which also accommodates the separate and unconnected steam generators. Fuel is a uranium-zirconium alloy with 13% U enrichment (or U-Pu-Zr with 11% Pu) with a 15-year life. After this the module is removed, stored on site until the primary lead (or Pb-Bi) coolant solidifies, and it would then be shipped as a self-contained and shielded item. A new fuelled module would be supplied complete with primary coolant. The ENHS is designed for developing countries but is not yet close to commercialisation.

A related project is the Secure Transportable Autonomous Reactor - **STAR** being developed by DOE's Argonne National Laboratory. It is a fast neutron modular reactor cooled by lead-bismuth eutectic, with passive safety features. Its 300-400 MWt size means it can be shipped by rail and cooled by natural circulation. It uses U-transuranic nitride fuel in a 2.5 m diameter cartridge which is replaced every 15 years. Decay heat removal is by external air circulation. The **STAR-LM** was conceived for power generation, running at 578°C and producing 180 MWe.

STAR-H2 is an adaptation for hydrogen production, with reactor heat at up to 800°C being conveyed by a helium circuit to drive a separate thermochemical hydrogen production plant, while lower grade heat is harnessed for desalination (multi-stage flash process). Any commercial electricity generation then would be by fuel cells, from the hydrogen. Its development is further off.

A smaller STAR variant is the Small Sealed Transportable Autonomous Reactor - **SSTAR**, being developed in collaboration with Toshiba and others in Japan (see 4S above). It has lead or Pb-Bi cooling, runs at 566°C and has integral steam generator inside the sealed unit, which would be installed below ground level. Conceived in sizes 10-100 MWe, main development is now focused on a 45 MWt/ 20 MWe version as part of the US Generation IV effort. After a 20-year life without refuelling, the whole reactor unit is then returned for recycling the fuel. The core is one metre diameter and 0.8m high. SSTAR will eventually be coupled to a Brayton cycle turbine using supercritical carbon dioxide. Prototype envisaged 2015.

For all STAR concepts, regional fuel cycle support centres would handle fuel supply and reprocessing, and fresh fuel would be spiked with fission products to deter misuse. Complete burnup of uranium and transuranics is envisaged in STAR-H2, with only fission products being waste.

Generation IV fast reactors

In 2003 the Generation IV International Forum (GIF) representing ten countries announced the selection of six reactor technologies which they believe represent the future shape of nuclear energy. These were selected on the basis of being clean, safe and cost-effective means of meeting increased energy demands on a sustainable basis, while being resistant to diversion of materials for weapons proliferation and secure from terrorist attacks. They will be the subject of further development internationally. Led by the USA, Argentina, Brazil, Canada, France, Japan, South Korea, South Africa, Switzerland, and the UK are members of the GIF, along with the EU.

Most of the six systems employ a closed fuel cycle to maximise the resource base and minimise high-level wastes to be sent to a repository. Three of the six are fast reactors and one can be built as a fast reactor, one is described as epithermal - these five are described below. Only two operate with slow neutrons like today's plants.

Of the five, only one is cooled by light water, one is helium-cooled and the others have lead-bismuth, sodium or fluoride salt coolant. The latter three operate at low pressure, with significant safety advantage. The last has the uranium fuel dissolved in the circulating coolant. Temperatures range from 510°C to 850°C, compared with less than 330°C for today's light water reactors, and this means that three of them can be used for thermochemical hydrogen production.

The sizes range from 150 to 1500 MWe (or equivalent thermal) , with the lead-cooled one optionally available as a 50-150 MWe "battery" with long core life (15-20 years without refuelling) as replaceable cassette or entire reactor module. This is designed for distributed generation or desalination.

At least four of the five systems have significant operating experience already in most respects of their design, which may mean that they can be in commercial operation well before 2030.

However, it is significant that to address non-proliferation concerns, the fast neutron reactors are not conventional fast breeders, ie they do not have a blanket assembly where plutonium-239 is produced. Instead, plutonium production takes place in the core, where burn-up is high and the proportion of plutonium isotopes other than Pu-239 remains high. In addition, reprocessing the fuel will enable recycling without separating the plutonium.

In February 2005 five of the participants signed an agreement to take forward the R&D on the six technologies. The USA, Canada, France, Japan and UK agreed to undertake joint research and exchange technical information.

While Russia was not initially a part of GIF, one design corresponds with the BREST reactor being developed there, and Russia is now the main operator of the sodium-cooled fast reactor for electricity - another of the technologies put forward by the GIF:

Gas-cooled fast reactors. Like other helium-cooled reactors which have operated or are under development, these will be high-temperature units - 850°C, suitable for power generation, thermochemical hydrogen production or other process heat. For electricity, the gas will directly drive a gas turbine (Brayton cycle). Fuels would include depleted uranium and any other fissile or fertile materials. Used fuel would be reprocessed on site and all the actinides recycled repeatedly to minimise production of long-lived radioactive wastes.

While General Atomics worked on the design in the 1970s (but not as fast reactor), none has so far been built. The French Atomic Energy Commission (CEA) is well advanced in design.

Lead-cooled fast reactors. Liquid metal (Pb or Pb-Bi) cooling is by natural convection. Fuel is depleted uranium metal or nitride, with full actinide recycle from regional or central reprocessing plants. A wide range of unit sizes is envisaged, from factory-built "battery" with 15-20 year life for small grids or developing countries such as teh SSTAR described above, to modular 300-400 MWe units and large single plants of 1400 MWe. Operating temperature of 550°C is readily achievable but 800°C is envisaged with advanced materials and this would enable thermochemical hydrogen production.

This corresponds with Russia's BREST fast reactor technology which is lead-cooled and builds on 40 years experience of lead-bismuth cooling in submarine reactors. Its fuel is U+Pu nitride. Initial development work is focused on two pool-type reactors: SSTAR - Small Secure Transportable Autonomous Reactor of 20 MWe in USA ,and the European Lead-cooled SYstem (ELSY) of 600 MWe in Europe.

Sodium-cooled fast reactors. This builds on some 390 reactor-years experienced with fast neutron reactors over five decades and in eight countries. It utilises depleted uranium in the fuel and has a coolant temperature of 550°C enabling electricity generation via a secondary sodium circuit, the primary one being at near atmospheric pressure. Three variants are proposed: a 50-150 MWe type with actinides incorporated into a U-Pu metal fuel requiring electrometallurgical processing (pyroprocessing) integrated on site, a 300-1500 MWe pool-type version of this, and a 600-1500 MWe type with conventional MOX fuel and advanced aqueous reprocessing in central facilities elsewhere.

In 2008 France, Japan and the USA signed two agreements to collaborate on developing sodium-cooled fast reactors. These were initially focused on using Phenix until it shut down in 2009, then on Japan's Monju, and they extend to aspects of fuel cycle. The work will involve demonstrating transmutation in connection with the Global Actinide Cycle International Demonstration (GACID) program, led by France. Beyond using Monju, the French CEA, the Japan Atomic Energy Agency and the US DOE have been discussing the size of planned prototypes, reactor types, fuel types, and schedules for deployment. The CEA has begun design of a prototype sodium fast reactor of 250 to 600 MWe, SFR, planned to operate in 2020.

Supercritical water-cooled reactors. This is a very high-pressure water-cooled reactor which operates above the thermodynamic critical point of water to give a thermal efficiency about one third higher than today's light water reactors from which the design evolves. The supercritical water (25 MPa and 510-550°C) directly drives the turbine, without any secondary steam system. Passive safety features are similar to those of simplified boiling water reactors. Fuel is uranium oxide, enriched in the case of the open fuel cycle option. However, it can be built as a fast reactor with full actinide recycle based on conventional reprocessing. Most research on the design has been in Japan.

Molten salt reactors. While not strictly a fast neutron reactor, the uranium fuel is dissolved in the sodium fluoride salt coolant which circulates through graphite core channels to achieve some moderation and an epithermal neutron spectrum. Fission products are removed continuously and the actinides are fully recycled, while plutonium and other actinides can be added along with U-238. Coolant temperature is 700°C at very low pressure, with 800°C envisaged. A secondary coolant system is used for electricity generation, and thermochemical hydrogen production is also feasible.

During the 1960s the USA developed the molten salt breeder reactor as the primary back-up option for the conventional fast breeder reactor and a small prototype was operated. Recent work has focused on lithium and beryllium fluoride coolant with dissolved thorium and U-233 fuel. The attractive features of the MSR fuel cycle include: the high-level waste comprising fission products only, hence shorter-lived radioactivity; small inventory of weapons-fissile material (Pu-242 being the dominant Pu isotope); low fuel use (the French self-breeding variant claims 50kg of thorium and 50kg U-238 per billion kWh); and safety due to passive cooling up to any size.

INPRO

As well as the GIF, another program with similar aims is coordinated by the IAEA. This is the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO). It was launched in 2001 and has 22 members including Russia, aiming "to support the safe, sustainable, economic and proliferation-resistant use of nuclear technology to meet the global energy needs of the 21st century." It does this by examining issues related to the development and deployment of Innovative Nuclear Energy Systems (INS) for sustainable energy supply.

One of the case studies in phase 1 of INPRO was undertaken by Russia on its BN-800 fast reactor, though the emphasis was on the methodology rather than the technology. Nevertheless, fast reactor systems will feature in further INPRO work.

US Advanced Fuel Cycle Initiative (AFCI)

AFCI incorporates earlier US work with the Integral Fast Reactor (IFR) project and international work on fast reactors. In 2006 it was rolled into the Global Nuclear Energy Partnership (GNEP), but then moved out of it in 2009. GNEP's main thrust was to counter proliferation concerns, but would have the effect of much greater resource utilisation as well.

GNEP envisaged fabrication and leasing of fuel for conventional reactors, with the used fuel being returned to fuel supplier countries and pyro-processed to recover uranium and actinides, leaving only fission products as high-level waste. The actinide mix would then be then burned in on-site fast reactors. The fast reactor and reprocessing aspects of this program continue in the USA under AFCI.

Innovative designs

An old design which has resurfaced as the travelling wave reactor (TWR) has been considered in the past as generically, a candle reactor, since it burns slowly from one end of a core to the other. The reactor uses depleted uranium packed inside hundreds of hexagonal pillars. In a “wave” that moves through the core at only one centimetre per year, the U-238 is bred into Pu-239, which is the actual fuel and undergoes fission. The reaction requires a small amount of enriched uranium to get started and could run for decades without refueling. However it is a low-density core and needs to be relatively large. The reactor uses liquid sodium as a coolant, core temperatures are about 550ºC. In 2009 this was selected by MIT Technology Review as one of ten emerging technologies of note. In 2010 the company promoting it, Terrapower, made overtures to Toshiba concerning its development.

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