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8

Fast Breeder Reactor Programs: History and Status

**Thomas B. Cochran, Harold A. Feiveson,
Walt Patterson, Gennadi Pshakin, M.V. Ramana,
Mycle Schneider, Tatsujiro Suzuki, Frank von Hippel**

Research Report 8
International Panel on Fissile Materials

Fast Breeder Reactor Programs: History and Status

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About the IPFM

The International Panel on Fissile Materials (IPFM) was founded in January 2006. It is an independent group of arms-control and nonproliferation experts from seventeen countries, including both nuclear weapon and non-nuclear weapon states.

The mission of the IPFM is to analyze the technical basis for practical and achievable policy initiatives to secure, consolidate, and reduce stockpiles of highly enriched uranium and plutonium. These fissile materials are the key ingredients in nuclear weapons, and their control is critical to nuclear disarmament, halting the proliferation of nuclear weapons, and ensuring that terrorists do not acquire nuclear weapons.

Both military and civilian stocks of fissile materials have to be addressed. The nuclear weapon states still have enough fissile materials in their weapon stockpiles for tens of thousands of nuclear weapons. On the civilian side, enough plutonium has been separated to make a similarly large number of weapons. Highly enriched uranium is used in civilian reactor fuel in more than one hundred locations. The total amount used for this purpose is sufficient to make about one thousand Hiroshima-type bombs, a design potentially within the capabilities of terrorist groups.

The Panel is co-chaired by Professor R. Rajaraman of Jawaharlal Nehru University in New Delhi and Professor Frank von Hippel of Princeton University. Its members include nuclear experts from Brazil, China, France, Germany, India, Ireland, Japan, South Korea, Mexico, the Netherlands, Norway, Pakistan, Russia, South Africa, Sweden, the United Kingdom and the United States. Professor José Goldemberg of Brazil stepped down as co-chair of IPFM on July 1, 2007. He continues as a member of IPFM.

IPFM research and reports are shared with international organizations, national governments and nongovernmental groups. It has full panel meetings twice a year in capitals around the world in addition to specialist workshops. These meetings and workshops are often in conjunction with international conferences at which IPFM panels and experts are invited to make presentations.

Princeton University's Program on Science and Global Security provides administrative and research support for the IPFM.

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1 Overview: The Rise and Fall of Plutonium Breeder Reactors

Frank von Hippel

The possibility of a plutonium-fueled nuclear reactor that could produce more fuel than it consumed (a "breeder reactor") was first raised during World War II in the United States by scientists in the atomic bomb program. In the following two decades, the Soviet Union, the United Kingdom, France, Germany, Japan and India followed the United States in establishing national plutonium breeder reactor programs, while Belgium, Italy and the Netherlands joined the French and German programs as partners. In all of these programs, the main driver was the hope of solving the long-term energy supply problem using the large scale deployment of nuclear energy for electric power. Plutonium-fueled breeder reactors appeared to offer a way to avoid a potential shortage of the low-cost uranium required to support such an ambitious vision using other kinds of reactors.

Uranium proved to be much more abundant than originally imagined and, after a fast start, nuclear power growth slowed dramatically in the late 1980s and global nuclear capacity is today about one-tenth the level that had been projected in the early 1970s. The urgency of deploying fast-neutron reactors for plutonium breeding therefore abated — at least in the western Organization for Economic Co-operation and Development (OECD) countries. In India and Russia, however, concerns about potential near-term uranium shortages persist, and new demonstration breeder reactors are being built. China, which currently is building up its nuclear capacity at an enormous rate, is considering the possibility of building two Russian-designed breeder reactors. Because of the high costs and reliability and safety issues that are detailed below, however, no commercial breeder reactors have been deployed.

Interest in fast-neutron reactors persists in the OECD countries for a new reason, political difficulties with storing or disposing of spent fuel. "Reprocessing" spent fuel does not eliminate the problem of siting a geological repository but a reprocessing plant does provide an interim destination that has proved a path forward with regard to the spent fuel problem in a number of nations.

Spent-fuel reprocessing was originally launched in countries that planned to deploy breeder reactors. They wanted separated plutonium for manufacturing startup fuel for their first breeder reactors. Standard light-water-reactor spent fuel contains about one percent plutonium. In the absence of breeder reactors, the separated plutonium has become a disposal problem and some countries have decided to recycle it into fuel for the same reactors that produced it. Slow-neutron reactors are relatively ineffective, however, in fissioning some of the plutonium isotopes, which therefore build up in recycled fuel.

Fast-neutron-reactor advocates argue that, if the plutonium and other long-lived transuranics in spent fuel could be fissioned almost entirely, the political problem of finding a geological disposal site for radioactive waste consisting of mostly shorter-lived fission products would become much easier. Fast neutron reactors would be more effective in fissioning all the transuranic isotopes. Fast-neutron breeder reactors could be converted into transuranic "burner" reactors by removing the plutonium-breeding uranium blankets around their cores and flattening the cores into more of a "pancake" shape so that more neutrons would leak out of them.

	MWe	MWt	Operation
France			
Rapsodie		40	1967–83
Phénix	250		1973–2009
Superphénix	1240		1985–98
India			
FBTR		40	1985–
PFBR	500		2010?
Japan			
Joyo		140	1977–
Monju	280		1994–95, 2010?
USSR/Russia			
BR-5		5	1959–2004
BOR-60	12		1969–

	MWe	MWt	Operation
USSR/Russia (cont.)			
BN-350 (Kazakhstan)	350		1972–99
BN-600	600		1980–
BN-800	800		2014?
United Kingdom			
DFR	15		1959–77
PFR	250		1974–94
United States			
EBR-I	0.2		1951–63
EBR-II	20		1963–94
Fermi 1	66		1963–72
SEFOR		20	1969–72
Fast Flux Test Facility		400	1980–93

Table 1.1 **Major experimental, pilot and demonstration fast breeder reactors.**[1]

This report looks at the experience and status of breeder reactor programs in France, India, Japan, the Soviet Union/Russia, the United Kingdom and the United States. The major breeder reactors built in these countries are listed in table 1.1. Germany also built two breeder reactors. All were sodium cooled.

The problems described in the country case studies in the following chapters make it hard to dispute Admiral Hyman Rickover's summation in 1956, based on his experience with a sodium-cooled reactor developed to power an early U.S. nuclear submarine, that such reactors are "expensive to build, complex to operate, susceptible to prolonged shutdown as a result of even minor malfunctions, and difficult and time-consuming to repair."[2]

Fast-neutron breeder reactors

Fissile isotopes are the essential nuclear materials in both nuclear reactors and nuclear weapons. They undergo fission when they absorb neutrons and, on average, release more neutrons than they absorb. This makes a sustained chain-reaction possible in a "supercritical mass." This supercritical mass must contain a significant concentration of fissile isotopes and must be large enough so that only a small fraction of the neutrons escape without interacting.

The most important fissile materials are uranium-235 and plutonium-239. Uranium-235 is found in nature, constituting 0.7 percent of natural uranium. Plutonium-239 is created when uranium-238 (99.3 percent of natural uranium) absorbs a neutron (figure 1.1).
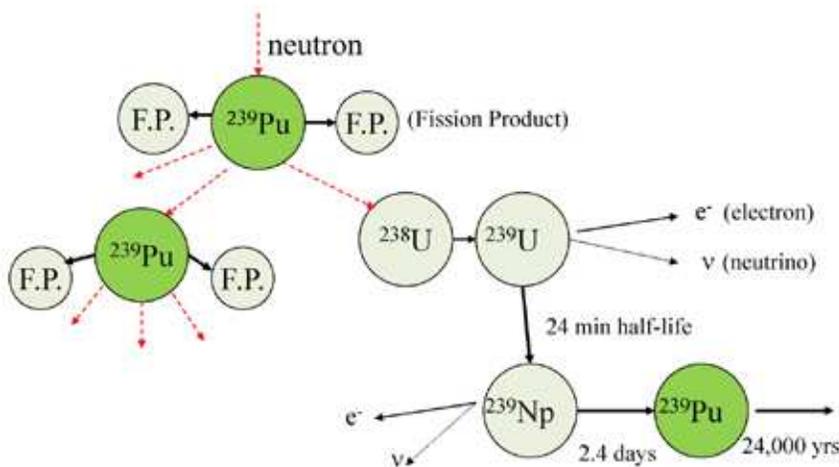

Figure 1.1 **Plutonium breeding.** A plutonium breeder reactor produces more plutonium than it consumes by using its extra fission neutrons to convert uranium-238 to uranium-239, which changes by radioactive decays involving electron and neutrino emission into neptunium-239 and then plutonium-239.

The vast majority of deployed power reactors around the world are fueled with low-enriched uranium and use a neutron "moderator" — in most cases, ordinary water, which also serves as the reactor coolant — that slows the neutrons and increases the likelihood that they will be captured by uranium-235 and cause it to fission. Such reactors are called "light-water reactors" and are fueled by uranium typically enriched to four–five percent in uranium-235. Light-water reactors are so named to distinguish them from the "heavy-water reactors" developed by Canada, which are fueled by natural uranium. In both types of reactors, the neutrons lose most of their energy in collisions with hydrogen, ordinary hydrogen in light-water reactors and heavy-hydrogen or deuterium in heavy-water reactors. In both types of reactors, some of the extra neutrons from uranium-235 fissions are also captured by uranium-238, converting it into chain-reacting plutonium-239 — but not enough to replace the fissioned uranium-235.

Virtually all breeder reactor programs have focused on reactors that do not use water as a coolant, so that the neutrons propagating the chain-reaction remain energetic (fast).

In order to be supercritical with fast neutrons, the "cores" of breeder reactors contain over 20 percent of fissile material — usually plutonium-239 — mixed with natural or "depleted uranium" (the residue after uranium-235 has been extracted from natural uranium by uranium-enrichment plants). Surrounding this core on all sides is a "blanket" — usually also consisting of natural or depleted uranium. Neutrons that leak out of the core are absorbed by the uranium-238 in the blanket and convert it into plutonium. Because such a reactor produces more plutonium than it consumes, its ultimate fuel is uranium-238, which is 140 times more abundant than uranium-235.

Plutonium breeder reactor programs have focused on fast-neutron reactors because, when a fast neutron fissions a plutonium-239 nucleus, more secondary neutrons are produced per fission than with any other combination of neutron speed and fissile isotope.[3] Fast-neutron plutonium-fueled reactors can therefore breed extra fissile material more rapidly than any other reactor system. Despite the safety, cost and reliability issues of fast-neutron reactors, this fact determined their choice as the preferred technology at a time when the global population of nuclear power reactors was expected to double every decade indefinitely. The extra plutonium produced by fast-neutron reactors could be used to provide startup fuel for additional plutonium breeder reactors, allowing the number of breeder reactors to grow at a high rate.

In Russia, given the cost and safety problems associated with plutonium fuel thus far, demonstration fast-neutron reactors have been fueled with highly-enriched uranium, enriched to between 20 and 30 percent uranium-235.

Why commercialization of breeder reactors failed

The rationale for pursuing breeder reactors — sometimes explicit and sometimes implicit — was based on the following key assumptions:

1. Uranium is scarce and high-grade deposits would quickly become depleted if fission power were deployed on a large scale;

2. Breeder reactors would quickly become economically competitive with the light-water reactors that dominate nuclear power today;

3. Breeder reactors could be as safe and reliable as light-water reactors; and,

4. The proliferation risks posed by breeders and their "closed" fuel cycle, in which plutonium would be recycled, could be managed.

Each of these assumptions has proven to be wrong.

Uranium is cheap and abundant. Breeder reactors were seen as a solution for the uranium scarcity problem because, by converting uranium-238 into chain-reacting plutonium, they can potentially increase one-hundred-fold the amount of fission energy that can be extracted from a kilogram (kg) of uranium and make it economically feasible to mine much lower grades of uranium ore.[4]

In 2007, uranium requirements for the global fleet of nuclear power reactors were 67,000 metric tons — approximately 180 tons per gigawatt of generating capacity per year. The International Atomic Energy Agency (IAEA) projects that global nuclear capacity will increase and that uranium requirements will increase correspondingly to between 94,000 and 122,000 tons a year in 2030.[5]

In 2008, the biennial report put out by the OECD Nuclear Energy Agency, *Uranium 2007: Resources, Production and Demand* — also known as "the Red Book" — found that, despite inflation, global known conventional resources of uranium recoverable for less than $130/kg had increased from about 4.7 to about 5.5 million tons. The Red Book also reported estimates from 27 countries that, with further exploration, an additional 7.6 million tons of uranium would be discovered in the same cost range.[6] At $130/kg, the cost of uranium would contribute 0.3 U.S. cents to the cost of a kilowatt-hour of nuclear electricity.

In the long run, worldwide, the amount of uranium recoverable at low cost is virtually certain to be far greater than the numbers reported in the Red Book. If plausible estimates of geological abundance are used, the amount of uranium still to be discovered at recovery costs up to $130/kg would be 50–126 million tons.[7] This corresponds to 500 to 1000 times the projected demand in 2030.

It will be seen from figure 1.2 that the price of uranium on the spot market went significantly above $130/kg during the late 1970s and then again after 2005. Except for these two periods when there was disequilibrium between supply and demand, prices have been less than $50 per kg. The 1970s price peak was due

to the expectation of an enormous expansion in nuclear power capacity. This expectation was not realized but large stockpiles of uranium were built up and then sold off during the subsequent decades resulting in the closure of many uranium mines. The sale by Russia to the U.S. of low-enriched uranium blended down from 500 tons of weapon-grade uranium from excess Cold War weapons at a rate sufficient to fuel half of the U.S. nuclear capacity extended the period of low demand for freshly mined uranium.[8] The stockpiles of natural uranium have been largely used up, however, and the blend-down of the Russian weapon-grade uranium will be completed in 2013. The most recent uranium price peak therefore reflected, at least in part, the expectation, compounded by speculation, that there might be uranium shortages before uranium-mining capacity increases again to the level required to support growing demand.

In any case, unlike the situation with oil or gas-fueled power plants, the cost of uranium fuel can double without having a significant impact on the cost of nuclear power. As noted above, at $130/kg, the cost of uranium contributes only 0.3 cents to the cost of a kilowatt-hour (kWh), which is about 5 percent of the cost of electricity produced by a new light-water reactor.[9]

Breeder reactors are costly to build and operate. Governments of countries in the OECD have together reported that they have spent about $50 billion (2007$) on fission and breeder reactor research and development (figure 1.3). Of this total, the United States reported that it had spent $15 billion, Japan $12 billion, the United Kingdom $8 billion, Germany $6 billion and Italy $5 billion. France
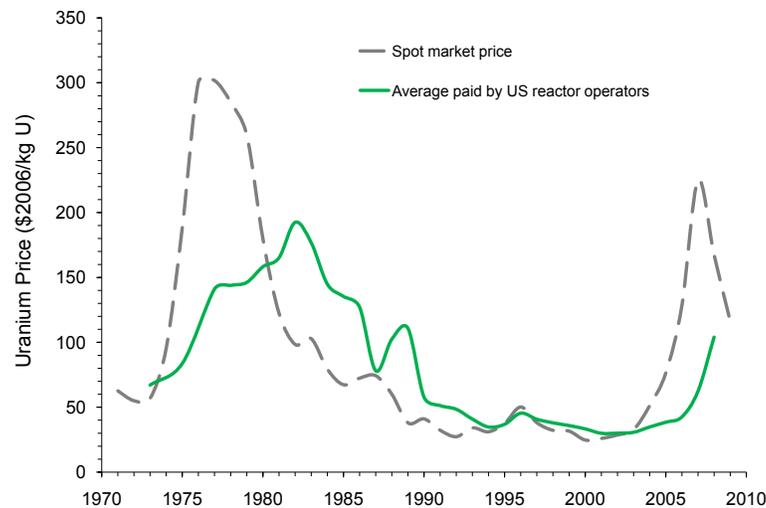
Figure 1.2 **History of the price of uranium since 1970.**[10]
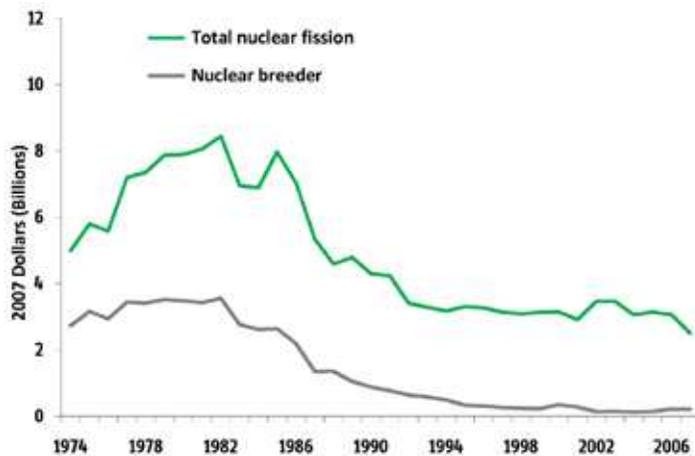
Figure 1.3 **Total fission and breeder research, development and demonstration funding in the OECD countries that had substantial breeder programs (1974 to 2007);**[11] Belgium, France, Germany, Italy, Japan, Netherlands, the United Kingdom and the United States.

reported only $1 billion in expenditures but this was obviously an incomplete report, given that the total cost of the *Superphénix* demonstration project alone is estimated at FRF 65 billion (1998FRF) or $14 billion (2007$) (see chapter 2).

Russia and India, which are both outside the OECD, have spent large amounts on breeder research, development and demonstration. The Soviet Union and Russia alone have spent an estimated $12 billion (see chapter 5). Yet none of these efforts has produced a reactor that is anywhere near economically competitive with light-water reactors.

The individual country studies make clear that, without astronomically high uranium prices, breeder reactors are unlikely to be economically competitive with light-water reactors. For "demonstration" liquid-sodium-cooled reactors the capital costs per kilowatt (KW) generating capacity have typically been more than twice those of water-cooled reactors of comparable capacity. Since breeder reactors were never built in quantity, it could be expected that, in production, this cost ratio would decline. Few if any argue today, however, that the capital costs for breeder reactors could be less than 25 percent higher than for water-cooled reactors of similar generating capacities. This would be a capital cost difference on the order of $1000 per kilowatt of generating capacity. At a 10 percent capital charge and a 90 percent average capacity factor, this would translate to a cost difference of about 1.3 cents per kilowatt hour.

Detailed economic comparisons of light-water reactors and breeder reactors using different breeding ratios, fuel reprocessing and fabrication costs, and capital costs show that direct disposal of spent light-water-reactor fuel would be far less expensive than reprocessing and plutonium recycle in breeder reactors under a wide range of assumptions.[12]

Fast-neutron reactors have special safety problems. As already noted, fast-neutron reactors cannot use water as a coolant because collisions with the hydrogen nuclei in water quickly remove most of the kinetic energy from the neutrons. Also, in order to sustain a chain-reaction with fast neutrons, the fissile material in a reactor core must be more concentrated. As a result, fast-neutron-reactor cores are smaller than those of light-water reactors with the same power. This necessitates the use of a coolant that can efficiently carry away the heat. The coolant that has been used in all demonstration breeder reactors to date is a liquid metal that melts at relatively low temperatures — sodium.

Sodium has both safety advantages and disadvantages compared to water. Its primary safety advantage is that the reactor operates below the boiling point of liquid sodium (883 °C) and therefore at low pressure. By contrast, water-cooled reactors operate at high pressures — over 150 atmospheres for pressurized water reactors. Therefore, if there is a large break in a pipe of a water-cooled reactor, the water flashes into steam, leaving the reactor's intensely hot fuel without coolant unless the core is flooded with emergency cooling water. In the case of a sodium-cooled reactor, however, unless the break is below the top of the core, the sodium will continue to cover the core and absorb heat.

Sodium's major disadvantage is that it reacts violently with water and burns if exposed to air. The steam generators, in which molten-sodium and high-pressure water are separated by thin metal, have proved to be one of the most troublesome features of breeder reactors. Any leak results in a reaction that can rupture the tubes and lead to a major sodium-water fire.

As the country studies detail, a large fraction of the liquid-sodium-cooled reactors that have been built have been shut down for long periods by sodium fires. Russia's *BN-350* had a huge sodium fire. The follow-on *BN-600* reactor was designed with its steam generators in separate bunkers to contain sodium-water fires and with an extra steam generator so a fire-damaged steam generator can be repaired while the reactor continues to operate using the extra steam generator. Between 1980 and 1997, the *BN-600* had 27 sodium leaks, 14 of which resulted in sodium fires (see chapter 5).

Leaks from pipes into the air have also resulted in serious fires. In 1995, Japan's prototype fast reactor, *Monju*, experienced a major sodium-air fire. Restart has been repeatedly delayed, and, as of the end of 2009, the reactor was still shut

down. France's *Rapsodie*, *Phénix* and *Superphénix* breeder reactors and the UK's *Dounreay Fast Reactor* (DFR) and *Prototype Fast Reactor* (PFR) all suffered significant sodium leaks, some of which resulted in serious fires.

Sodium also creates radiation problems. When it absorbs a neutron, ordinary sodium-23 becomes sodium-24, a gamma-emitting isotope with a 15-hour half-life. The sodium that cools the core therefore becomes intensely radioactive. To ensure that a steam-generator fire does not disperse radioactive sodium, reactor designers have inserted an intermediate sodium loop. The heat generated from the reactor is transferred to non-radioactive sodium through a sodium-sodium heat exchanger. The non-radioactive sodium delivers the heat to the steam generators. The extra sodium loops and associated pumps contribute to the high capital costs of breeder reactors.

Finally, light-water-cooled reactors have the critical safety characteristic that, if the water moderator is lost, the chain-reaction stops. It is impossible to sustain a chain-reaction in 4 to 5 percent enriched uranium without slowing the neutrons so that they are captured preferentially by uranium-235. In the absence of the water, the fast neutrons will be absorbed mostly in uranium-238 and the chain-reaction ends.

By contrast, in a fast-neutron reactor, the concentration of plutonium is high enough that it can sustain a chain-reaction even in the event of a coolant loss. Indeed, except for special core configurations, the reactivity will increase if the coolant is lost.[13] Furthermore, if the core heats up to the point of collapse, it can assume a more critical configuration and blow itself apart in a small nuclear explosion.[14] Whether such an explosive core disassembly could release enough energy to rupture a reactor containment and cause a Chernobyl-scale release of radioactivity into the environment is a major concern and subject of debate. (See chapter 3 for a discussion of this debate in India.)

Sodium-cooled reactors have severe reliability problems. The reliability of light-water reactors has increased to the point where, on average, they operate at about 80 percent of their generating capacity. By contrast, a large fraction of sodium-cooled demonstration reactors have been shut down most of the time that they should have been generating electric power. A significant part of the problem has been the difficulty of maintaining and repairing the reactor hardware that is immersed in sodium. The requirement to keep air from coming into contact with sodium makes refueling and repairs inside the reactor vessel more complicated and lengthy than for water-cooled reactors. During repairs, the fuel has to be removed, the sodium drained and the entire system flushed carefully to remove residual sodium without causing an explosion. Such preparations can take months or years.

In contrast, when a water-cooled reactor is shut down, the top of the pressure vessel can be removed and the reactor cavity that holds the pressure vessel can be flooded with water to provide shielding against the radioactivity of the fuel and the irradiated steel. Repairs can take place guided by underwater periscopes and video cameras.

The history of the world's only commercial-sized breeder reactor, France's *Superphénix*, is dominated by lengthy shutdowns for repairs (see chapter 2). *Superphénix* went critical and was connected to the grid in January 1986 but was shut down more than half of the time until operations ceased in December 1996. Its lifetime capacity factor — the ratio of the number of kilowatt-hours that it generated to the number it could have generated had it operated continually at full capacity — was less than 7 percent. The histories of Japan's *Monju* and the U.K.'s *Dounreay* and *Prototype Fast Reactors* and the U.S. *Enrico Fermi 1* demonstration breeder reactor power plants were similarly characterized by prolonged shutdowns (see chapters 4, 6 and 7). Russia's *BN-600* has experienced a respectable capacity factor but only because of the willingness of its operators to continue to operate it despite multiple sodium fires.

The fast-neutron reactor fuel cycle provides easy access to plutonium for weapons. All reactors produce plutonium in their fuel but breeder reactors require plutonium recycle, the separation of plutonium from the ferociously radioactive fission products in the spent fuel. This makes the plutonium more accessible to would-be nuclear-weapon makers. Breeder reactors — and separation of plutonium from the spent fuel of ordinary reactors to provide startup fuel for breeder reactors — therefore create proliferation problems.

This fact became dramatically clear in 1974, when India used the first plutonium separated for its breeder reactor program to make a "peaceful nuclear explosion." Breeders themselves have also been used to produce plutonium for weapons. France used its *Phénix* breeder reactor to make weapon-grade plutonium in its blanket. India, by refusing to place its breeder reactors under international safeguards as part of the U.S.-India nuclear deal, has raised concerns that it might do the same.

India's *Prototype Fast Breeder Reactor* (PFBR), expected to be completed in 2010, will have the capacity to make 90 kg of weapon-grade plutonium per year, if only the radial blanket is reprocessed separately and 140 kg per year if both radial and axial blankets are reprocessed.[15] The Nagasaki bomb contained 6 kg of weapon-grade plutonium and modern weapons designs contain less. At 5 kg per warhead, the *PFBR* would produce enough weapon-grade plutonium for 20–30 nuclear weapons a year, a huge increase in production capacity in the context of the South Asian nuclear arms race.

The G.W. Bush Administration proposed to make reprocessing more "proliferation resistant" by leaving some of the other transuranic elements (neptunium, americium and curium) mixed with the plutonium.[16] Even if all the transuranics
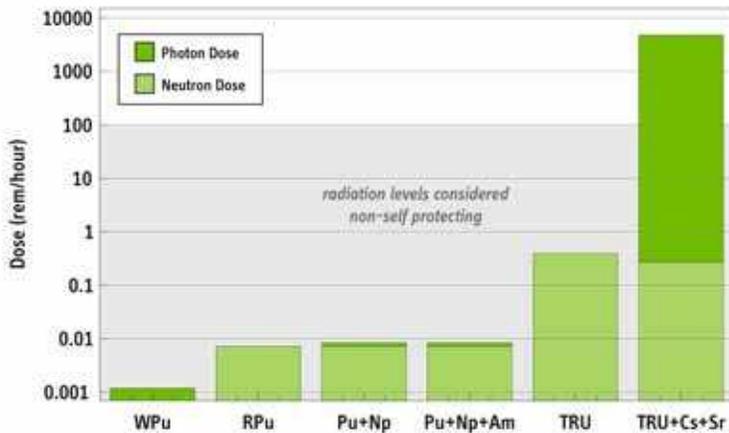
Figure 1.4 **Dose rate of a 4.4 kg container of various mixtures of separated transuranics compared to spent fuel.** Peak dose rates only approach the IAEA's self-protection standard (100 rem/hour at one meter distance) if high-activity fission products are included. Cesium-137, which has a half-life of 30 years, dominates the radiation field from spent fuel after ten years. (Weapon-grade plutonium [WPu], reactor-grade plutonium [RPu], neptunium [Np], americium [Am], transuranic waste [TRU], cesium [Cs] and strontium [Sr]).[17]

were left mixed with the plutonium, however — a project that the U.S. Department of Energy abandoned when it learned that the technology was not in hand — the gamma radiation field surrounding the mix would still be less than one-hundredth the level the IAEA considers self-protecting against theft and thousands of times less than the radiation field surrounding plutonium when it is in spent fuel (figure 1.4).

Prospects for breeder reactors
After six decades and the expenditure of the equivalent of tens of billions of dollars, the promise of breeder reactors remains largely unfulfilled and efforts to commercialize them have been steadily cut back in most countries.

Germany, the United Kingdom and the United States have abandoned their breeder reactor development programs. Despite the arguments by France's nuclear conglomerate Areva, that fast-neutron reactors will ultimately fission all the plutonium building up in France's light-water reactor spent fuel,[18] France's only operating fast-neutron reactor, *Phénix*, was disconnected from the grid in March 2009 and scheduled for permanent shutdown by the end of that year.[19] The *Superphénix,* the world's first commercial-sized breeder reactor, was abandoned in 1998 and is being decommissioned. There is no follow-on breeder reactor planned in France for at least a decade.

Japan's *Monju* reactor operated for only a year before it was shut down by an accident in 1995 and it had not resumed operation as of the end of 2009. There are plans for a new demonstration reactor by 2025 and commercialization of breeder reactors by 2050 but there is reason to doubt these projections. Japan's Government is not willing to kill its breeder program entirely, because, as in France, the breeder is still the ultimate justification for Japan's spent fuel reprocessing program. For decades, however, the Japanese Government has been reducing funding for its breeder program and shifting commercialization further and further into the future (see chapter 4).

Russia and India are building demonstration breeder reactors. In both cases, however, their breeder (and spent fuel reprocessing) programs leave much to be desired regarding the availability of data on reliability, safety and economics. In the case of India, there is also the potential for use of breeder reactors to produce plutonium for weapons. The high costs of commercial breeder reactors and an international Fissile Material Cutoff Treaty that bans production of fissile materials for weapons will force some of these issues into the open and foster new debates about the value of these breeder programs.

In the United States, during the G.W. Bush Administration, fast reactors returned to the agenda as "burner" reactors. In an initiative started in 2006 labeled "The Global Nuclear Energy Partnership (GNEP)," the U.S. Department of Energy proposed that sodium-cooled fast-neutron reactors be used to make the radioactive waste in spent reactor fuel more manageable. With the removal of the uranium blankets around their cores, fast-neutron reactors would, like light-water reactors, breed less fissile material than they burned. The high-energy neutron spectrum of the sodium-cooled reactors would be more effective, however, in fissioning the non-chain-reacting isotopes of plutonium and minor transuranic elements. Already in 1996, however, a National Academy of Sciences assessment commissioned by the U.S. Department of Energy, had concluded that such an effort would have very high costs and marginal benefits and would take hundreds of years of recycling to reduce the global inventory of transuranic isotopes by 99 percent.[20] The Obama Administration and the U.S. Congress share this skepticism and propose a new research and development program to investigate alternative strategies for managing U.S. spent fuel.[21]

The breeder reactor dream is not dead but it has receded far into the future. In the 1970s, breeder advocates were predicting that the world would have thousands of breeder reactors operating by now. Today, they are predicting commercialization by approximately 2050. In the meantime, the world has to deal with the legacy of the dream; approximately 250 tons of separated weapon-usable plutonium and ongoing — although, in some cases struggling — reprocessing programs in France, India, Japan, Russia and the United Kingdom.

Endnotes

[1]   Adapted from the respective chapters in this report and from "Fast Neutron Reactors," World Nuclear Association, <http://www.world-nuclear.org/info/inf98.html> (accessed 15 September 2009).

[2]   Richard G. Hewlett and Francis Duncan, *Nuclear Navy: 1946–1962* (Chicago: University of Chicago Press, 1974): 274.

[3]   Another breeding cycle with a lower maximum "breeding ratio" of fissile material produced over fissile material consumed but which can operate with slow neutrons is based on thorium-232, which is even more abundant than uranium-238, and is converted by neutron capture into the chain-reacting isotope, uranium-233.

[4]   For a breeder reactor, even the three grams of uranium in a ton of average crustal rock, if fissioned completely, would release almost ten times as much energy as is contained in a ton of coal. Specifically, if fully fissioned, 3 grams of uranium would release approximately 3 megawatt-days of heat or 260×10^9 joules. The combustion energy of a ton of coal is approximately 30×10^9 joules.

[5]   Nuclear Energy Agency, Organization for Economic Co-operation and Development, *Uranium 2007, Resources Production and Demand* (Vienna: OECD Publishing, 2008), 11, table 27. OECD nuclear capacity is estimated to increase only modestly, from 309 to 312–375 gigawatts electric in 2030. Non-OECD capacity is assumed to grow much more; from 61 to 200–288 gigawatts electric in 2030.

[6]   Nuclear Energy Agency, Organization for Economic Co-operation and Development, op. cit., tables 1, 2, 4, 11.

[7]   Erich Schneider and William Sailor, "Long-term Uranium Supply Estimates," *Nuclear Technology* 162 (2008): 379.

[8]   The United States has been buying 30 tons per year of 90 percent enriched uranium after it has been blended down to low-enriched uranium with 1.5 percent blend-stock produced from Russia's stocks of depleted uranium. If blended down to 4.4 percent enrichment, 916 tons of low enriched uranium would be produced each year. A light-water reactor, operating at an average of 90 percent of its nominal generating capacity, would consume approximately 18.6 tons of this low-enriched uranium per year.

[9]   Assuming 0.25 percent uranium-235 remaining in the depleted uranium, 9 kg of natural uranium is required to produce 1 kg of uranium enriched to 4.4 percent. At that enrichment, a kg of uranium can release 53 megawatt-days of fission heat in a light-water reactor. Assuming a conversion efficiency of one third of that heat to electric energy, 47,000 kWh of electricity would be generated per kg of uranium. Enrichment, fabrication, and operations and maintenance charges should be added to the 0.3 cents per kWh charge for uranium. At 0.25 percent tails, enriching natural uranium to 4.4 percent uranium-235 will require 6.6 separative work units/kg. At $140/separative work unit (SWU), this would add approximately 0.25 cents per kWh. Fabrication would cost approximately $250/

kg corresponding to less than 0.1 cents per kWh. Operations and maintenance would add approximately 1.5 cents/kWh. Finally, for a light-water reactor costing $4000/KWe operating at a 90 percent capacity factor the capital charge would be 5 cents/kWh, assuming a 10 percent capital charge; Massachusetts Institute of Technology, *The Future of Nuclear Power, An MIT Interdisciplinary Study* (Cambridge: MIT Press, 2003), Appendix 5A.

[10]   This figure is based on figure 5 of Matthew Bunn, Steve Fetter, John Holdren and Bob van der Zwaan, "The Economics of Reprocessing Versus Direct Disposal of Spent Nuclear Fuel," *Nuclear Technology* 150 (June 2005): 209. It has been updated by Steve Fetter through 2006 and the author through 2007 (average U.S. price) and 2008 and early 2009 (spot price) based on U.S. Energy Information Administration, "Average Price and Quantity for Uranium Purchased by Owners and Operators of U.S. Civilian Nuclear Power Reactors by Pricing Mechanisms and Delivery Year," <http://www.eia.doe.gov/cneaf/nuclear/umar/table5.html> (accessed 15 September 2009); and *Uranium Intelligence Weekly* respectively.

[11]   From International Energy Agency R&D Statistics Database, <http://www.iea.org/Textbase/stats/rd.asp> (accessed 15 September 2009). Unavailable values have been replaced with zeros.

[12]   Matthew Bunn et al., op. cit., 209. For example, for utility financing at a 10 percent discount rate, central values for reprocessing and breeder core fabrication of $1000 and $1500 per kg of heavy metal respectively, and only a small capital cost difference between light-water reactors and breeders of $200/KWe (5 percent of current light-water reactor capital costs), the breakeven uranium price would be $340 per kg — far greater than projected prices even if nuclear power grows substantially in the coming decades.

[13]   Since the sodium slows the neutrons somewhat, its removal increases reactivity since both the fission probability of plutonium-239 and the number of neutrons released per fission increase with neutron energy. The only way to offset this positive reactivity feedback from loss of coolant is to design the core geometry so that leakage of neutrons out of the fuel region of the core increases as the sodium is lost. This requires either that the core be pancake shaped or that neutron absorbing blanket fuel assemblies be dispersed among the fuel assemblies.

[14]   H.A. Bethe and J.H. Tait, "An Estimate of the Order of Magnitude of the Explosion When the Core of a Fast Reactor Collapses," UKAEA-RHM 56 (1956).

[15]   Alexander Glaser and M.V. Ramana, "Weapon-grade Plutonium Production Potential in the Indian Prototype Fast Breeder Reactor," *Science and Global Security* 15 (2007): 85–105.

[16]   The National Energy Policy Development Group, "Report of Vice President Cheney's 2001 Energy Task force," 5–21. See also U.S. Department of Energy, Office of Nuclear Energy, Science, and Technology, "Report to Congress on Advanced Fuel Cycle Initiative: The Future Path for Advanced Spent Fuel Treatment and Transmutation Research", (2003).

17 Robert Hill, Argonne National Laboratory, "Advanced Fuel Cycle Systems: Recycle/Refabrication Technology Status," September 2005. See also, E.D. Collins, Oak Ridge, "Closing the Fuel Cycle can Extend the Lifetime of the High-Level-Waste Repository," American Nuclear Society, 2005 Winter Meeting, Washington, DC; and, Jungmin Kang and Frank von Hippel, "Limited Proliferation Resistance Benefits from Recycling Unseparated Transuranics and Lanthanides from Light-water Reactor Spent Fuel," *Science and Global Security* 13 (2005): 169.

18 Areva is reprocessing France's low-enriched uranium fuel and recycling the plutonium into mixed-oxide (MOX, uranium-plutonium) fuel for light-water reactors. The spent MOX fuel contains approximately two thirds as much plutonium as the fresh MOX fuel but the plutonium contains an increased fraction of the even isotopes, plutonium-238, plutonium-240 and plutonium-242, that are difficult to fission in thermal reactors. The spent MOX fuel therefore is being stored in the hopes that fast-neutron reactors will eventually be built that can fission all the plutonium isotopes efficiently.

19 Mycle Schneider, Steve Thomas, Antony Froggatt and Doug Koplow, "World Nuclear Industry Status Report 2009," (August 2009), 101.

20 Committee on Separations Technology and Transmutation Systems, National Research Council, *Nuclear Wastes: Technologies for Separation and Transmutation* (Washington, D.C: National Academies Press, 1996).

21 In June 2009, the U.S. Department of Energy announced that it was cancelling development of the Global Nuclear Energy Partnership Programmatic Environmental Impact Statement because it was no longer pursuing domestic commercial reprocessing.

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2 Fast Breeder Reactors in France

Mycle Schneider

France's program to produce and separate plutonium began immediately after the Second World War. While the initial purpose was to obtain plutonium for the nuclear weapons program, very early on the fast breeder reactor became a second strategic goal. European cooperation was another goal and the EUROCHEMIC consortium was created in 1957 with the participation of 10 countries; France and Germany held the largest shares with 17 percent each.[1]

The first reprocessing plant, the "plutonium factory" (usine de plutonium, UP1), began operating in Marcoule in 1958 and the first proposal for the experimental fast reactor *Rapsodie* was drawn up that year. Preliminary studies for a 1000 megawatt electric (MWe) reactor were conducted as early as 1964.

The behavior of materials was tested under neutron irradiation in *Harmonie* starting in 1965 and breeder core configurations were studied in the critical facility, *Masurca*, starting in 1966. These research facilities were located at the Cadarache site in southern France. Much later, in 1982, the *Esmeralda* facility, also at Cadarache, was designed to study sodium fires. While most of the research was financed by the French Atomic Energy Commission (CEA), up to 35 percent of some research projects were funded by EURATOM.

In 1966, the second commercial reprocessing plant UP2, financed entirely by the CEA (with the civil and military budgets paying equal shares), started operations at La Hague by separating plutonium from gas-graphite reactor fuel. In Belgium, the EUROCHEMIC plant began operating in 1967. It operated until 1974 and reprocessed 181.3 tons of spent fuel of various types and origins. Two years later the CEA started up a light-water reactor head-end at La Hague (UP2-400) and launched the 100 percent daughter company COGEMA under private law. Foreign (German) light-water reactor fuel was sent to La Hague as early as 1973. There had been no experience with reprocessing light-water reactor fuel with much higher burn-ups than gas-graphite reactor fuel and it took COGEMA eleven years, until 1987, to operate at a nominal capacity of 400 tons per year.

A version of this chapter has been published in *Science and Global Security* 17 (2008): 36-53.

Rapsodie, Cadarache

Construction of France's first experimental sodium-cooled reactor, *Rapsodie*, started in 1962 and it went critical on 28 January 1967 with a nominal capacity of 20 megawatts thermal (MWt). At the end of 1967, its power was increased to 24 MWt, and in 1970, after core redesign, to 40 MWt. Its operating power was reduced to 22 MWt in June 1980 to minimize the thermal stresses thought to be the source of cracks in the reactor vessel. The reactor operated until April 1983, when it was shut down permanently.

Rapsodie was a loop-type reactor, with the heat exchanger between the primary and secondary sodium loops outside the reactor vessel. It was as close as possible to the basic design imagined for commercial applications (molten-sodium coolant, reactor material, power density, etc). The core contained 31.5 kilograms (kg) of plutonium-239 and 79.5 kg of uranium-235. The mean duration of reactor runs was 80 days and the fuel reached burn-ups of 102,000 MWd/t.[2]

Phénix, Marcoule

In February 1968, when *Rapsodie* had been operating for one year, excavation work began at Marcoule for the construction of the 250 MWe (563 MWt) *Phénix* reactor. In 1969, the CEA and Electricité de France (EDF, France's Government-owned utility) signed a protocol for the joint construction and operation of the *Phénix* plant. Ownership and costs were shared 80 percent by the CEA and 20 percent by EDF. The standard *Phénix* core contains 931 kg plutonium containing 77 percent plutonium-239. The reactor went critical on 31 August 1973 and was connected to the grid on 13 December 1973,[3] a year ahead of the 250 MWe *Prototype Fast Reactor* (*PFR*) in the United Kingdom. Until 2005, the mean length of reactor runs was 90 days and the fuel reached burn-ups of up to 150,000 MWd/t.[4]

On 17 October 1973, between the dates of criticality and grid connection of *Phénix*, OPEC member countries halted oil deliveries to a number of countries that supported Israel and significantly increased the price of crude oil. In 1974, the French Government committed to its first large series of power reactors, 16 units. The International Atomic Energy Agency (IAEA) forecast up to 4,450 gigawatts (GW) of nuclear power installed by year 2000. Between 1973 and 1976 uranium prices increased from $6 to $40 per pound on the spot market. Plutonium was seen as a solution to long-term nuclear fuel supply concerns.

Until the end of the 1980s, *Phénix* had a remarkable operational record. Then, after a number of unexplained reactivity transients, the load factor plunged virtually to zero. The incidents had serious potential safety implications. The reactor remained shut down most of the period between 1991 and 1994 until an extensive research program was carried out. It was restarted for very short periods, however — probably to avoid the legal requirement of an entire new licensing procedure after a two-year shutdown. In addition, a costly refurbishment program was undertaken between 1994 and 2002 (see figure 2.1 for operational history).
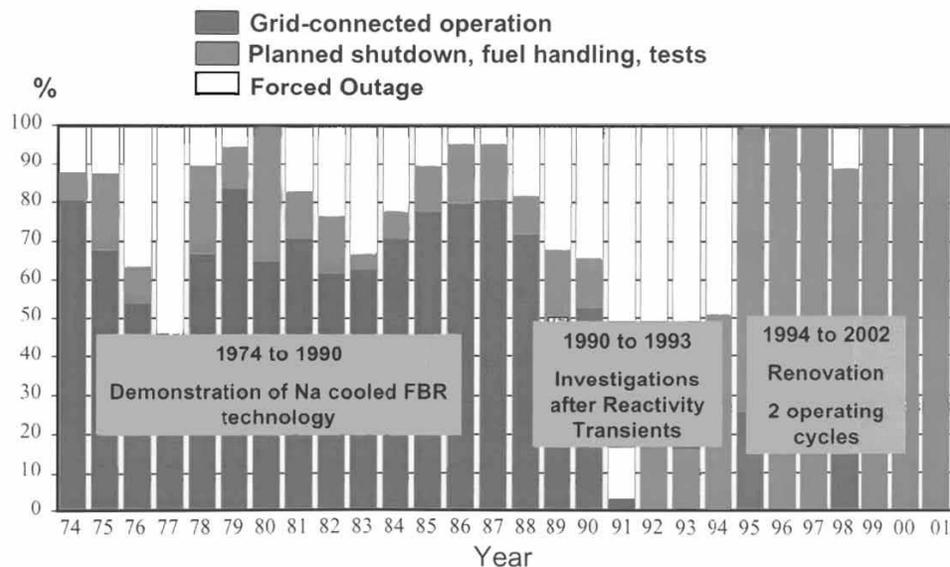
Figure 2.1 Operational history of France's Phénix breeder reactor, 1974–2002.
Na is the chemical symbol for sodium. Source: IAEA, Fast Reactor Database 2006 Update.

In June 2003, the National Safety Authority ASN (Autorité de Sûreté Nucléaire) authorized the restart of *Phénix* for six refueling periods at less than two thirds of its original power. This allowed operation until the end of 2008 and into 2009. Nominal power was decreased from 233 megawatt electric (MWe) net to 130 MWe net. As of the end of 2007 the reactor had a cumulative load factor of 44.6 percent.[5] *Phénix* was shut down in 2009.

Superphénix, Creys-Malville

In 1971 and 1972, even prior to the first oil shock, utilities from France, Germany and Italy signed a number of agreements for joint construction of two commercial breeder reactors, one in France and one in Germany. In December 1972 the French Parliament passed a law that granted permission to create companies "that carry out an activity of European interest in the electricity sector".[6] The legislation was tailor-made for the creation of a European fast-neutron reactor consortium (NERSA),[7] which was established in 1974, shortly after the start-up of *Phénix*, with the purpose of building the first commercial-size plutonium-fueled fast breeder reactor in the world.[8] The Superphénix Parliamentary Enquiry Committee later noted that the "public enquiry into the project was excessively short." It lasted only a month from 9 October to 8 November 1974.[9]

The project immediately attracted significant opposition. In November 1974, 80 physicists of the Lyon Physics Institute highlighted specific risks of breeder technology and, in February 1975, approximately 400 scientists initiated an appeal that detailed their concerns about France's nuclear program in general and the fast breeder in particular. That same year, the German utility RWE transferred its NERSA shares to the European consortium, SBK, that planned to build the SNR-300 breeder reactor in Kalkar, Germany.[10] André Giraud, then head of CEA, urged the rapid and massive introduction of breeders, since delays in their introduction would have "catastrophic consequences on the uranium savings that are expected."[11] The public enquiry commission into the *Superphénix* project estimated that fast breeders would supply a quarter of France's nuclear electricity by the year 2000.

In the middle of April 1976, the Restricted Energy Council chaired by President Valery Giscard d'Estaing made the political decision to build *Superphénix*. Site preparation work started immediately at Creys-Malville (45 km East of Lyon, 60 km from Grenoble and 70 km from Geneva). The Parliamentary Enquiry Committee noted 22 years later:

> Once the decision to build was taken, the electricity utilities would not rest until they succeed. Convinced of the well founded decision, they did not allow local consultation to slow them down; the latter can be qualified as minimal.[12]

The official public decision to build *Superphénix* was only announced a year later. The Parliamentary Enquiry Committee wonders:

> Finally, what to think of a governmental decision to authorize the creation of the plant dated 12 May 1977, thus taking place after the beginning of the preliminary infrastructure and site preparation work and after the beginning of the construction of the reactor?[13]

In the summer of 1976 some 20,000 people occupied the site to protest the construction of *Superphénix*. Around 50 municipalities in the region had come out in opposition to the project between 1974 and 1976 and, in November 1976, about 1300 scientists from the Geneva region issued an open letter to the Governments of France, Italy, Germany and Switzerland voicing their concerns over the project.

CEA Chairman and soon to be named Minister of Industry André Giraud was more optimistic than ever and, at the December 1976 meeting of the American Nuclear Society in Washington D.C., forecasted 540 commercial breeders in the world for the year 2000, of which 20 would be in France. By 2025, he projected the number of *Superphénix*-size fast breeder reactors units worldwide would reach exactly 2766.[14] In fact, not a single *Superphénix*-size fast breeder reactor was in operation in the world in 2000.

On 31 July 1977, a large international demonstration close to the construction site in Creys-Malville, with some 50,000 participants, turned extremely violent. The riot police used grenades that led to the death of Vital Michalon, a local teacher. Another demonstrator lost a foot and a third had a hand amputated. The events were a profound trauma for the French anti-nuclear movement. The State did not alter its plans. Three days after the events, René Monory, then Minister, of Industry, declared: "The Government will continue the construction at Creys-Malville and *Superphénix*, because it is a matter of life and comfort of the French people."[15] The construction proceeded.

The combination of the EURODIF uranium enrichment consortium that started up its plant at Tricastin in 1979 and the push for a European plutonium industry were attempts to acquire independence from what some decision makers and industry leaders perceived as U.S. nuclear supremacy. France's President Giscard d'Estaing declared that "if uranium from French soil is used in fast breeder reactors, we in France will have potential energy reserves comparable to those of Saudi Arabia."[16] U.S. President Jimmy Carter's non-proliferation policy, highly critical of plutonium separation and use, was considered "totally absurd" by the CEA.[17]

In 1982, Jean-Louis Fensch, a CEA engineer, produced a 250 page report on fast breeders for the Superior Council on Nuclear Safety, a consultative body. Fensch concluded that "fast breeder reactors are the most complicated, the most polluting, the most inefficient and the most ambiguous means that man has invented to date to reduce the consumption of nuclear fuel".[18]

By the time *Superphénix* went critical in 1985, international enthusiasm for nuclear power had already peaked and the number of construction starts in the world had gone down from a peak of 40 units in 1975 to 13 in 1985 and 1 in 1986.[19] The Chernobyl catastrophe in 1986 only accelerated the decline in nuclear projects. *Superphénix*, whose objective was to save uranium, was outdated by the time it began operating. Uranium prices had dropped from $40 to $15 per pound on the spot market, little more than the 1974 price. In comparison with the demand, uranium resources were abundant.

France's nuclear decision makers did not alter their plans, however. The result was that the country built up both a large electric-power generating overcapacity (at least a dozen excess nuclear units by the middle of the 1980s) and a full-scale plutonium economy that had long lost its raison d'être. Between 1987 and 1997 the rate of reprocessing of spent fuel at La Hague quadrupled to almost 1700 tons per year, of which approximately half was for foreign clients. With an approximate one percent content of plutonium, the La Hague facilities separated about 17 tons of plutonium in 1997. This was roughly the magnitude of the total cumulated quantity of plutonium that had been irradiated in French breeder reactors as of the end of 1996 when *Superphénix* was permanently shut down.[20]
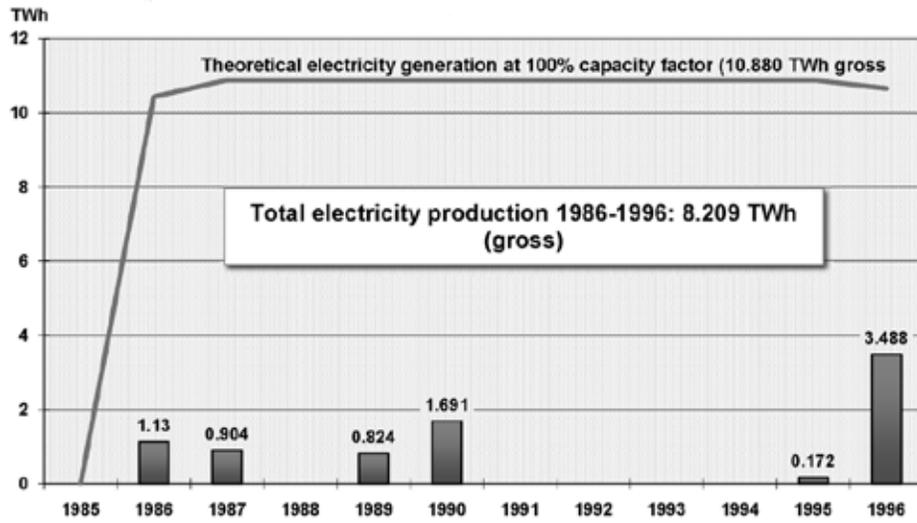
Figure 2.2 **Superphénix annual electricity generation.** Source: CEA, WISE-Paris.
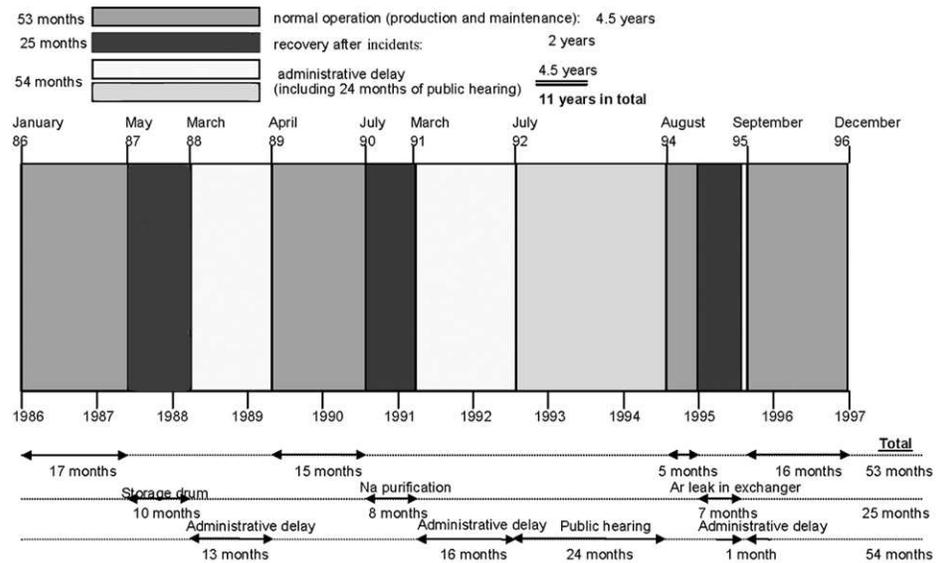

Figure 2.3 **Superphénix operational and administrative history.**
Source: IAEA, Fast Reactor Database 2006 Update.

The core of *Superphénix* contained 5780 kg of plutonium (4054 kg of plutonium-239). Operated at a nominal capacity with annual one-third core refueling, *Superphénix* would have absorbed over 1900 kg of plutonium per year. But during its 11 years of operations, the reactor did not even use the equivalent of one reactor core.

Superphénix had a rated power of 1200 MWe net (1240 MWe gross). On 7 September 1985 it went critical and was connected to the grid on 14 January 1986. It was plagued by a number of technical and administrative problems, however, and was shut down more than half of the time until 24 December 1996 when it produced its last kilowatt hour (kWh). *Superphénix* generated 8.2 terawatt hours (TWh) (gross) in total, almost half of which was generated during its last year of operation. Its lifetime load factor was less than 7 percent.

As figures 2.2 and 2.3 illustrate, *Superphénix* experienced a series of significant incidents and administrative hurdles. The reactor never operated more than 17 months in a row. Operations halted in May 1987 with the discovery of a major sodium leak in the fuel transfer tank or storage drum. The tank could not be repaired and it took 10 months to develop a new method to load and discharge fuel from the reactor core.

The incident also revealed major deficiencies in the French fast breeder reactor organization. Before the leak, at the end of 1985, FRAMATOME's engineering subsidiary NOVATOME laid off more than half of its staff, 430 of 750 employees. NOVATOME was losing a lot of money because it could not invoice NERSA for work on *Superphénix* until it had gone into commercial operation.[21] In the course of the relocation of its thinned-out engineering teams from Paris to Lyon, many experts took up attractive offers to leave NOVATOME. As a result, when the storage tank leak occurred, NERSA realized that the specialist who had managed the electronic database for the tank had left the organization and it took some time before the database could be accessed. The re-qualification and authorization of the new fuel transfer and storage method absorbed another 13 months before the reactor could restart in April 1989. Low-power operation lasted until July 1990 when a defective compressor led to major air leakage into the system and oxidation of the sodium. Sodium purification took another eight months. In December 1990, the roof of the turbine hall collapsed after a heavy snowfall (figures 2.4 and 2.5).

On 3 June 1991, NERSA requested permission to restart the reactor by July 1991. On 27 May 1991, however, the French Conseil d'Etat invalidated the 1989 restart license that had been legally challenged by Swiss and French opponents. The restart, unlike the original licensing procedure, became subject to a lengthy process of parliamentary hearings and debates on a national and regional level. In June 1992, the Government decided to commission expert reports and to request a new public enquiry that was carried out between 30 March and 14 June 1993. The public enquiry commission issued its report on 29 September 1993 and the safety authorities reported to the Government in January 1994. A new operating license was finally issued on 11 July 1994. The unit had been back on line for only seven months, however, when an argon leak in a heat exchanger forced a new outage. When the reactor restarted in September 1995, it was for the last time.

Figure 2.4 **Superphénix turbine hall in foreground.**
Photo: Dissident-Media.

Figure 2.5 **Superphénix collapsed turbine hall roof.**
Photo: Dissident-Media.

On Christmas 1996, *Superphénix* was shut down for maintenance, core reconfiguration and the launch of a research program into transmutation. On 28 February 1997, however, the Conseil d'Etat nullified the July 1994 operating permit and, on 19 June 1997, incoming Prime Minister Jospin told the National Assembly that "*Superphénix* will be abandoned." The political decision became official on 2 February 1998 when the communiqué of an inter-ministerial committee meeting stated that "the Government has decided that *Superphénix* will not restart, not even for a limited period of time".

A Green Party representative had entered a European National Government with a senior ministerial position for the first time. Dominique Voynet became Environment Minister, and thereby shared oversight over civil nuclear safety in France with the Industry Minister. Point number one on the Green Party electoral platform had been the closing of *Superphénix*. The issue had always been highly symbolic for France's nuclear power opponents. It would have been difficult to imagine anything less than the end of the *Superphénix* project after the Green Party joined the Government. It is also perfectly clear, however, that at least part of EDF's top management had long considered *Superphénix* and reprocessing a costly error.[22]

French diplomats were quick to downplay the strategic significance of the end of *Superphénix*. The French Embassy in the U.S. stated in its "Nuclear Notes from France":[23]

> In the wake of recent decisions, made by the French Government, including the closure of the *Superphénix* fast breeder reactor, some may wonder if France is changing its nuclear policy. Basically, the answer is no. Both Prime Minister Lionel Jospin and Economic Minister Dominique Strauss-Kahn have made it clear France is satisfied with its nuclear "wise" commitment, stressing the large return on investment it provides in terms of economic competitiveness, self-sufficiency and environmental protection. France will stick to its policy of reprocessing and plutonium recycling, a good way to optimize waste management while producing more electricity. Is it surprising? Just remember what everybody in France has in mind: no oil, no gas, and no coal means no choice! It sometimes helps!

A decree dated 30 December 1998 formalized the decision to proceed with the final closure of *Superphénix* and the first decommissioning steps. As of 2008, the fuel has been discharged and transferred to the storage facility APEC on site. The turbine hall has been emptied. A permit for full decommissioning was issued on 20 March 2006.

Military plutonium from Phénix

The CEA's military department had a keen interest in fast breeders because of the fact that, as a by-product, they generate super-grade plutonium in the breeder blankets.[24] Even if the utilities involved in the *Superphénix* project always categorically rejected the idea of a military link, it is clear that *Phénix* was used for the generation of plutonium for France's nuclear-weapon program. The potential militarization of *Superphénix* raised considerable concern, especially in Germany, and was discussed in the context of the possibility that France might develop and deploy neutron bombs in Europe.[25]

In the case of *Phénix*, the fuel design allowed not only for the use of the radial blanket but also part of the axial blanket to produce plutonium for weapons. Usually the axial blanket is integrated with the core fuel in the same fuel pins but it seems that in the case of *Phénix* the upper axial blanket was separate. *Phénix* blanket material was reprocessed at the military UP1 plant in Marcoule, while core material, diluted with gas-graphite reactor fuel, was reprocessed at La Hague and at a dedicated pilot plant at Marcoule (APM with the head end SAP-TOP, later SAP-TOR).

In unusually blunt statement, General Jean Thiry, former director of the French nuclear test sites in the Sahara and in the Pacific, who prior to these positions had been responsible for eight years for plutonium "counting" at the CEA, told the daily *Le Monde* in 1978: "France is able to make nuclear weapons of all kinds and all yields. It will be able to fabricate them in large numbers as soon as the fast breeder reactors provide it with abundant quantities of the necessary plutonium."[26] In 1987 General Thiry confirmed his statement and declared: "One can always get plutonium, especially if one develops... This is apparently an idea that one should not say (openly) because it is not moral,[27] but I defend Creys-Malville (*Superphénix*) and the fast breeder reactor type, because there you have plutonium of extraordinary military quality."[28] Dominique Finon states that *Phénix* was used for military purposes starting in 1978 but that the idea to use *Superphénix* for defense needs was abandoned in 1986.[29]

Research and development, construction, operation and decommissioning costs

France's fast breeder reactor program was costly to the French taxpayer. A comprehensive historical economic assessment is not available. An extensive analysis to the middle of the 1980s was carried out and the national Court of Auditors provided a cost estimate in 1996.[30] In addition a number of assessments have looked at specific aspects (R&D, decommissioning, etc.). Figure 2.6 provides an overview of *Phénix* operating costs between 1972 and 2003.

Between 1973 and 1996 the CEA alone spent an undiscounted FRF 15.8 billion ($2008 3.8 billion) on breeder R&D, 50 percent more than on light-water reactors (including the EPR development).[31]

According to an agreement signed in 1969, the CEA provided 80 percent and EDF 20 percent of the construction and operational costs of *Phénix*. Construction costs totaled FRF1974 800 million ($2008 880 million). Approximately €600 million ($2008 950 million) were spent on *Phénix* upgrades between 1997 and 2003.

The French state spent some FRF1985 44 billion ($2008 17.4 billion) on the fast breeder program between 1960 and 1986. The *Superphénix* construction costs increased by 80 percent to reach FRF1985 26 billion ($2008 9.5 billion) by the time the reactor went on line in 1986.[32] At that time, the investment cost ratio per installed kilowatt (KW) between breeder and PWR was evaluated by the CEA at 2.58.[33]
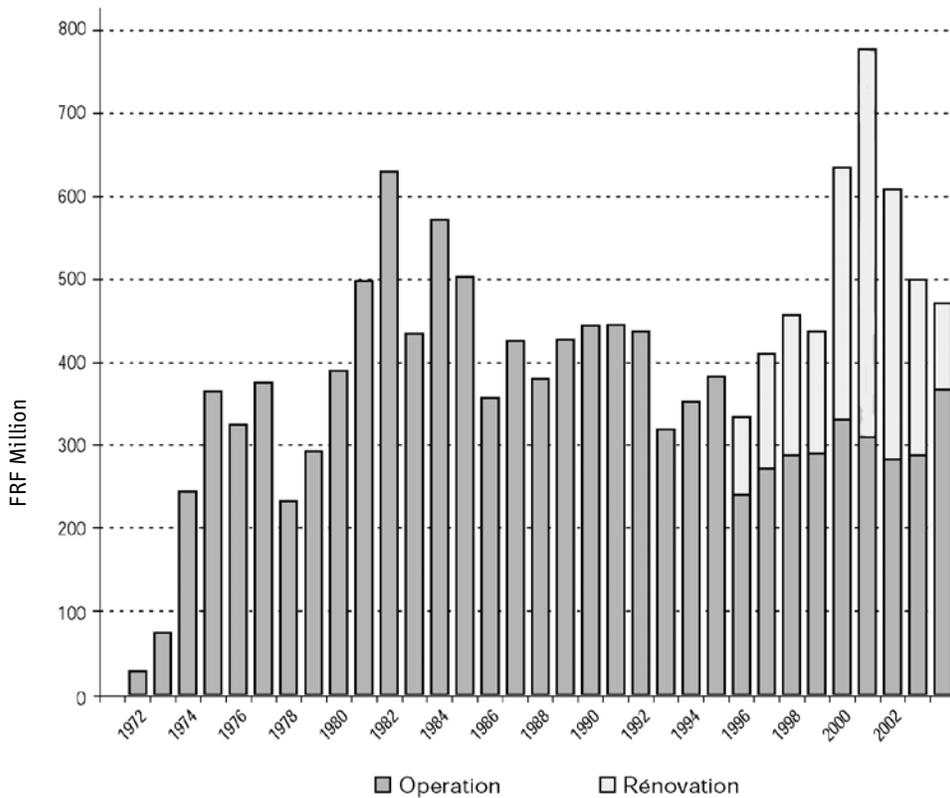

Figure 2.6 **Phénix operating costs, 1972–2003 (FRF2000 million).**
Source: Sauvage, 2004.

The Court of Auditors, in its 1996 annual report, provided an evaluation of the cost of *Superphénix*, assuming that it would operate until the end of 2001. It estimated that the unit had cost FRF 34.4 billion by the end of 1994 and that financial, spent fuel management, decommissioning and waste management costs would reach an additional FRF 27.4 billion. Operating costs were given at FRF 1.7 billion per year. Considering the fact that the unit shut down at the end of 1996, adding two years of operating costs but also of power generation (approximately 3.65 TWh), the total estimated cost would be somewhere around FRF 64 billion, minus approximately a FRF one billion electricity generation credit.[34] Jacques Chauvin, president of the directorate of NERSA stated that "in total, cumulating investment and operating costs and taking into account all future costs, *Superphénix* will have cost FRF 65 billion of which EDF will have paid 38 billion."[35]

The NERSA and Auditor Court figures are closer than the level of uncertainty attached. In particular, the decommissioning costs contain a substantial potential margin of error. They have been raised several times. As of 2003, the Court of Auditors estimated *Superphénix* decommissioning and waste management alone would cost €2.081 billion.

The Parliamentary Enquiry Committee concluded:

> In the end nobody seems to contest the judgment of the Court (of Auditors) that 'the record of the fast breeder experience appears unfavorable today in any case on the financial level'. Christian Pierret (Secretary of State for Industry) goes as far as qualifying it as 'unacceptable'.[36]

Safety problems in the French fast breeder reactor program

All three reactors, *Rapsodie*, *Phénix* and *Superphénix*, encountered significant safety problems during start-up, operation and dismantling periods; including sodium leaks, reactivity incidents, explosions and material failures.

Rapsodie – sodium leaks and a lethal explosion

After a rather smooth operational period from *Rapsodie*'s start-up at the beginning of 1967, at the end of 1978 a small primary sodium leak was detected, which led to the decision to reduce the operational capacity from 40 MWt to approximately 22 MWt. In January 1982, another small sodium leak was detected in the nitrogen system surrounding the primary vessel. Localization of the leak was believed to be too costly and too uncertain. The reactor was therefore shut down on 13 October 1982.

The secondary sodium was drained in April 1983 and is still stored on the Cadarache site. The primary sodium was drained by April 1984. It took two years to retrieve the 468 highly irradiated reflector assemblies from around the core

(222 made of nickel, 246 made of steel) from the vessel, wash them to eliminate traces of sodium, and install them in a storage container. The 37 tons of primary sodium were treated in a specially designed facility (DESORA) that turned it into 180 cubic meters of concentrated sodium hydroxide.

On 31 March 1994, an explosion occurred during the cleaning of the residual primary sodium contained in a tank located in a hall outside the containment building.[37] An experienced, highly specialized 59 year old CEA engineer was killed instantly and four people were injured. Approximately 100 kg of residual sodium had remained at the bottom of a tank at the end of the treatment campaign. An analysis of the accident concluded later:

> The process selected to perform this clean up operation consisted in progressively introducing in the tank a heavy alcohol called ethylcarbitol, while monitoring the reaction through temperature, pressure, hydrogen and oxygen measurements. The major cause of the accident was due to the formation of an heterogeneous physical-chemical environment, complex and multiphasic made of three basic components: alcohol, alcoholate and sodium. This environment turned out to be particularly favourable to the development of thermal decomposition reaction and/or catalytic exothermal reactions. Large quantities of gases (including hydrogen and light hydrocarbon compounds) were thus produced. Shortly after the last alcohol injection on 31 March, the phenomenon ran out of control, leading to a sudden rupture of the overpressurised tank, then to the explosion of the gases mixture blown out in the hall.[38]

Since this accident, the use of ethylcarbitol or other heavy alcohol has been forbidden in the treatment of sodium. But the circumstances of the accident are subject to an ongoing legal dispute. In 2001 an expert court-commissioned analysis accused the CEA, the IPSN (Institute for Nuclear Protection and Safety, predecessor of IRSN) and the safety authorities of "faults by imprudence, negligence and violation of safety obligations."[39] As of December 2009, there still is no published information indicating that there has been a final judgment.

Phénix – sodium leaks and reactivity spikes
As of 1988, *Phénix* had a cumulative average load factor of 60.5 percent. Operation was not without problems, however. The first fuel pin leak occurred in June 1975, secondary sodium leaks occurred in September 1974, March and July 1975 (approximately 20 liters each for the first two and 1 liter for the last). "Leakage generally led to the slow spontaneous combustion of this sodium in the insulation, without triggering fires external to the insulation."[40] Repair operations proved ineffective and valves in the three secondary systems were eventually replaced by diaphragms.

On 11 July 1976, a sodium leak occurred at the intermediate heat exchanger (between the primary and secondary sodium loops) that led to what was later labeled as the "first real sodium fire in the *Phénix* plant." The fire was extinguished manually. On 5 October 1976, another sodium fire broke out at an intermediate heat exchanger and was again manually brought under control. Figure 2.7 provides an illustration of the impact of a sodium fire at an unidentified date. A further sodium leak was identified in August 1977. Further secondary sodium leaks were identified in the 1980s, including incidents in March and November 1984, and in September 1988.

Figure 2.7 **Phénix heat exchanger with insulation removed after sodium fire.** Photo: Sauvage, 2004

In July 1978, two control rods showed a level of swelling that prevented normal extraction from their guide tubes. However, since the blocking was positioned above the insertion level during normal operation, the phenomenon was considered not to constitute an immediate safety issue.

In the first years no events directly impacted the steam generators. Steam generator failures, which can lead to violent sodium-water reactions are the most feared incidents in fast-neutron reactors. But various incidents took place in the steam generator environment, including four water leaks in the economizer-evaporator inlet of the steam generators between November 1975 and September 1976. The first cladding failure was detected in May 1979. It led to the "greatest release of fission gas (xenon-135) ever seen in the *Phénix* plant".

Between April 1982 and March 1983, sodium-water reactions in the reheater stages affected all three steam generators in at least four incidents. In the first event, on April 1982, approximately 30 liters of water leaked into the sodium and created a combustion flame that burned a hole in two tubes and damaged the reheater module's shell. The other three events apparently involved quantities of water limited to a few liters. These four sodium-water incidents resulted in a total of six months of outage and nine months of operation limited to two-thirds capacity.

The most costly and potentially most significant incidents were rapid reactivity transients in the core on three occasions in 1989 (6 and 24 August, 14 September) and on 9 September 1990. In spite of a research program costing hundreds of millions of francs, 200 person-years of work, and the elaboration of some 500 documents, the cause of the phenomenon was never conclusively identified.

The events were particularly worrying since following reactivity and power drops of 28 percent to 45 percent within 50 milliseconds, power actually increased above the original state of the reactor. The fear was that such an event could trigger a power excursion. The cause could possibly have been an argon gas bubble going through the core, but this hypothesis was never confirmed. Subsequent investigations revealed that similar events had taken place in April 1976 and June 1978 and that the explanation at the time (control rod slippage) was wrong.

Superphénix – sodium leaks and missile attacks

Safety concerns related to the operation of the *Superphénix* reactor were a key objection of the critics of the project from its very early stages. Over 5,000 tons of highly reactive sodium combined with several tons of highly toxic plutonium raised numerous safety issues. After the Chernobyl accident, which occurred only three months after connection of *Superphénix* to the grid, the question of the positive void coefficient inherent in the design, theoretically favoring power-excursion accidents, only increased the concerns of a number of scientist-critics. Safety concerns played a significant role in generating the opposition, including its most extreme forms.[41]

The first exceptional event took place at Creys-Malville before construction of the reactor was completed. A group of anti-nuclear activists succeeded in obtaining an RPG-7 (Rocket Propelled Grenade launcher) and eight warheads ("bonbons") from the German terrorist organization Rote Armee Fraktion (RAF) via the Belgian counterpart Cellules Communistes Combattantes (CCC). On 18 January 1982, five missiles were fired against the *Superphénix* construction site (three other pieces of ammunition had been discarded prior to the attack). There was little material damage but significant political and media attention. The authors of the attack were never caught until the confession of the key person, Chaïm Nissim, 22 years later.[42]

The internal incident database of the French Nuclear Safety Authorities only refers to a single event during the operational period of *Superphénix*: a sodium leak from the main fuel storage tank. The tank was a key element of the plant since it was intended to serve as a transfer and storage tank for new and spent fuel assemblies. The leak was detected on 3 April 1987 and led to a 10-month shutdown. Worse, it became evident that it would be impossible to repair the tank. The leak was determined to be the result of a design error (wrong material). An entirely new fuel loading and unloading scheme had to be developed. It is interesting to note that the original design of the transfer tank did not have double walls. The consequences of the leak would most likely have been much more dramatic if that design had been used.

The National Assembly's Enquiry Committee on *Superphénix* and the fast breeder reactor line also discussed the three previously mentioned significant events: the sodium pollution of July 1990, the turbine hall roof collapse of December 1990, and the argon gas leak in December 1994.[43]

At present the *Superphénix* reactor is undergoing various decommissioning operations. The dismantling of its reactor block is planned to begin in 2014 and continue for a period of eight years. The entire installation is to be dismantled by 2025.

After four decades of R&D, design and operation of LMFRs, with no imminent new breeder project, CEA, EDF and AREVA agreed in 2000 to preserve the breeder knowledge-base.[44]

Endnotes

1   The OECD countries are Germany, France, Belgium, Italy, Sweden, Austria, Denmark, Norway, Netherlands, Switzerland, Portugal, Turkey. Spain joined the consortium in 1959. Most of these countries had nuclear weapons programs or ambitions at some point, including Germany, Italy, Sweden and Switzerland.

2   *IAEA, Fast Reactor Database: 2006 Update*, IAEA-TECDOC Series No. 1531 (Vienna: International Atomic Energy Agency, 2006), < http://www-frdb.iaea.org/index.html> (accessed 14 June 2009).

3   Two months after the last U.S. reactor order that was not subsequently cancelled.

4   IAEA, op. cit.

5   IAEA, *Power Reactor Information System database (PRIS)*. According to the IAEA PRIS, not including the years 1997-2004, <http://www.iaea.org/cgi-bin/db.page.pl/pris.ophis.htm?country=FR&site=PHENIX&units=&refno=10&opyear=2007&link=HOT> (accessed 14 June 2009).

6   *Journal Officiel*, 23 December 1972.

7   Centrale Nucléaire Européenne à Neutrons Rapides, S.A.

8   A French Government decree dated 13 May 1974 authorized EDF to take part in the consortium. The original shareholders other than EDF (51 percent) were ENEL, Italy (33 percent) and RWE (Rheinisch-Westphälisches Elektrizitätswerk), Germany (16 percent).

9   Robert Galley, Christian Bataille, "Rapport Fait au Nom de la Commission D'enquête Sur *Superphénix* et la Filière Des Réacteurs À Neutrons Rapides," French National Assembly, 25 June 1998.

10 Schnellbrüterkernkraftwerksgesellschaft, 68.85 percent RWE; 14.75 percent SEP (consortium of Dutch utilities); 14.75 percent Electronucléaire (consortium of Belgian utilities); 1.65 percent CEGB (UK).

11 Quoted in Jean-Louis Fensch (CEA engineer), Conseil Supérieur de la Sûreté Nucléaire, "Finalités du Retraitement" (commissioned by Government appointed advisory group, unpublished annex to the First "Castaing Report"), October 1982.

12 Robert Galley, Christian Bataille, op. cit.

13 Ibid.

14 Conseil Supérieur de la Sûreté Nucléaire, op. cit. Quoted by Jean-Louis Fensch (CEA engineer).

15 Ibid.

16 *Time* magazine, 18 February 1980.

17 *Le Point magazine*, 19 September 1977, quoted in Fensch, op.cit.

18 Quoted by Jean-Louis Fensch (CEA engineer). Conseil Supérieur de la Sûreté Nucléaire, op. cit.

19 IAEA, "Nuclear Power Reactors in the World," Technical Data Series (April 1987).

[20] Mycle Schneider (Dir.), Xavier Coeytaux, "Recyclage Des Matières Nucléaires – Mythes Et Réalités", commissioned by Greenpeace France, WISE-Paris, May 2000.

[21] The status of commercial or industrial operation would be considered achieved if the reactor operated at >60 percent rated capacity at least for a month of which at least one week without interruption.

[22] The author was an advisor to the French Environment Minister's office from 1998 to 2002. In 1998, one of the top five directors of EDF approached the author with the request to communicate to Dominique Voynet the message that, while EDF obviously could not make any public statements, she should know that she had a lot of support inside EDF. While Voynet became the perfect scapegoat for the plutonium lobby, she was the secret ally of those within the nuclear establishment who wanted to terminate fast breeder reactor and plutonium activities.

[23] Embassy of France in the U.S., "Nuclear Notes from France *52,"* Jan, Feb, March 1998. See for example <http://www.iaea.org/inisnkm/nkm/ws/countries/france_development.html> (9 November 2009).

[24] The radial breeder elements and axial blanket contain uranium-238 that "breeds" plutonium-239 in very pure form. High-grade plutonium with a share of approximately 97 percent of fissile plutonium (plutonium-239 and plutonium-241) is particularly well suited for nuclear weapons.

[25] If France had decided to produce neutron bombs in large numbers, there is little doubt that it could not have generated enough weapon-grade plutonium without militarizing additional reactors besides the two Célestin heavy water reactors and *Phénix*.

[26] *Le Monde*, 19 January 1978.

[27] Today one would say "politically correct."

[28] Interview with the author, 18 June 1987, unpublished.

[29] Dominique Finon, "L'échec des Surgénérateurs, Autopsie d'un Grand Programme," PUG 1989, 76.

[30] Dominique Finon, op. cit.

[31] For details see Mycle Schneider (dir), "Research and Development Expenditure on Nuclear Issues in France 1960 – 1997," commissioned by Energy Services, Lohmar, Germany, WISE-Paris, February 1998.

[32] Dominique Finon, op. cit., footnote 84, 305. This sum includes financial costs of FRF 7.5 billion and engineering as well as pre-operational costs of some FRF 2 billion covered by EDF beyond the direct investment costs indicated by the CEA as FRF 16.5 billion.

[33] French Atomic Energy Commission (CEA), "Coûts de la Centrale *SuperPhénix*," Note BRPC 86.317, January 1986, quoted in Dominique Finon, op. cit.

[34] The Court of Auditors assumed an average price of FRF 0.25/kWh, which would

mean a credit of less than FRF 1 billion for the generation of the 3.65 TWh.

35 Robert Galley, Christian Bataille, op. cit.

36 Ibid.

37 Also the date of the final shutdown of the *PFR* in Dounreay.

38 G. Rodriguez et al., "General Review of the Decommissioning of Liquid Metal Fast Reactors (LMFRs) in France," in "Operational And Decommissioning Experiences With Fast Reactors," IAEA TECDOC, Series No. 1405 (Proceedings of a meeting on August 11-15, 2004 at CEA, Cadarache, France).

39 *Libération*, 15 March 2007.

40 Quotes in this section are from Jean-François Sauvage, *Phénix, 30 Years of History – The Heart of a Reactor,* (Bagnols-sur-Cèze: CEA-EDF, 2004). The book is available free of charge at <http://www.iaea.org/inisnkm/nkm/aws/fnss/phenix/book/index.html> (accessed 14 June 2009).

41 A positive void coefficient is when a bubble in the core increases its reactivity. For a more detailed description see Chapter 3, India and Fast Breeder Reactors.

42 C. Nissim*, L'amour et le Monster, Roquettes Contre Creys-Malville* (Lausanne, Paris: Favre, 2004). Chaïm Nissim, played a key role in the acquisition and firing of the rockets at the *Superphénix* construction site. Nissim has detailed the process of acquisition and attack in this book.

43 Robert Galley, Christian Bataille, op. cit.

44 F. Baque, "R&D LMFRs Knowledge Preservation French Project", paper presented in the IAEA Technical Meeting on Operational and Decommissioning Experience with Fast Reactors, Cadarache, France, 11–15 March 2002.

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3 India and Fast Breeder Reactors

M. V. Ramana

India is one of only two countries that are currently constructing commercial scale breeder reactors. (The other is Russia.) Both the history of the program and the economic and safety features of the reactor suggest, however, that the program will not fulfill the promises with which it was begun and is being pursued.

History

Breeder reactors in India were originally proposed in the 1950s as part of a three-stage nuclear program as a way to develop a large autonomous nuclear power program despite India's relatively small known resource of uranium ore.[1]

The first stage of the three-phase strategy involves the use of uranium fuel in heavy water-reactors, followed by reprocessing the irradiated spent fuel to extract the plutonium.

In the second stage, the plutonium is used to provide startup cores of fast breeder reactors. These cores would be surrounded by blankets of either depleted or natural uranium, to produce more plutonium. If the blanket were thorium, it would produce chain-reacting uranium-233. So as to ensure that there is adequate plutonium to construct follow-on breeder reactors, however, breeder reactors would have to be equipped with uranium blankets until the desired nuclear capacity was achieved.

The third stage would involve breeder reactors using uranium-233 in their cores and thorium in their blankets. Though the thorium-uranium-233 cycle would result in slow growth of nuclear power, presumably the rationale for going to this stage was to completely eliminate the requirement for uranium.

The three-stage program remains the official justification for pursuing breeders, despite their slow and disappointing progress.

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Though India's Department of Atomic Energy (DAE) has been talking about breeder reactors since its inception, work on even conceptual studies on breeders began only in the early 1960s. In 1965, a fast reactor section was formed at the Bhabba Atomic Research Center (BARC) and design work on a 10 MWe experimental fast reactor was initiated.[2] This seems to have been abandoned and, in 1969, the DAE entered a collaboration agreement with the French Atomic Energy Commission (CEA) and obtained the design of the *Rapsodie* test reactor and the steam generator design of the *Phénix* reactor.[3] This was to be the *Fast Breeder Test Reactor* (*FBTR*), India's first breeder reactor.

As part of the agreement with the CEA, a team of approximately thirty Indian engineers and scientists were trained at Cadarache, France. Once they returned, they formed the nucleus of the Reactor Research Centre (RRC) that was set up in 1971 at Kalpakkam to lead the breeder effort. In 1985, this was renamed the Indira Gandhi Centre for Atomic Research (IGCAR). Over the years, the center has emerged as the main hub of activities related to India's breeder program.

The Fast Breeder Test Reactor experience

The budget for the *FBTR* was approved by DAE as early as September 1971 and it was anticipated that the *FBTR* would be commissioned by 1976.[4] The reactor finally attained criticality only in October 1985 and the steam generator began operating in 1993.[5]

Much of the first one and a half decades of the *FBTR*'s operations were marred by several accidents of varying intensity. Two of these are described below in some detail to illustrate the complexities of dealing with even relatively minor accidents and the associated delays, as well as the hazards posed to workers. When viewed in combination with similar experiences elsewhere, these circumstances suggest that it is unlikely that sodium-cooled breeder reactors will ever perform with the reliability that water-cooled reactors have demonstrated over the past two decades.

In May 1987 there was a major incident that took two years to rectify.[6] This occurred as a fuel subassembly was being transferred from the core to the periphery.[7] The problem began with the failure of a protective circuit involved in the rotation of the plug to move the selected fuel assemblies. For some reason, this protective circuit was bypassed and the plugs were rotated with a foot long section of one fuel subassembly protruding into the reactor core. This resulted in the bending of that specific subassembly as well as the heads of 28 reflector subassemblies on the path of its rotation. Various maneuvers to rectify the situation did not help and only resulted in one reflector subassembly at the periphery getting ejected as well as the bending of a sturdy guide tube by 32 cm. The last event has been described as the result of "a complex mechanical interaction" which seems to suggest that how it happened was never really understood.

Extensive repairs were required before the reactor could be restarted. First, the guide tube had to be cut into two parts using a specially designed remote cutting machine while ensuring that none of the chips produced during the cutting process fell into the core.[8] Then the damaged reflector subassemblies had to be identified using a periscope. Finally part of the sodium had to be drained out and the damaged subassemblies removed using specially designed grippers. As might be expected, all of this took time and reactor operations commenced only in May 1989.[9]

The second accident described here is one that is common in fast breeder reactors – a sodium leak. That this occurred seventeen years after the reactor was commissioned underscores the generic nature of such accidents. The leak occurred in September 2002 inside the purification cabin, which houses the pipelines of the primary sodium purification circuit.[10] The cause of the leak is said to have been "the defective manufacturing process adopted in the manufacture of the bellows sealed sodium service valves". By the time the leak could be confirmed and controlled, approximately 75 kilograms of sodium had spilled over and solidified on the cabin floor and various components in that cabin.

Removing this radioactive sodium was a major effort. To begin with, even to approach the cabin, the workers had to wait ten days to allow for a reduction in the radioactivity from the sodium, some of which had absorbed a neutron to become Na-24, a gamma emitter (15-hour half-life). Even then, in areas near the spilled sodium, the dose rate was as high as 900 millisieverts per hour (mSv/h).[11] Another problem resulted from the whole cabin normally being surrounded by a layer of nitrogen so as to avoid sodium burning. At first, IGCAR tried to simply replace the nitrogen with regular air so that cleanup workers could breathe. But this led to sparks and fires involving the spilled sodium. These had to be put out with dry chemical powders – but then this led to lots of dust being suspended in the atmosphere and made visibility poor. Once again nitrogen had to be reintroduced. Workers were then sent in with masks that had tubes feeding them with breathing air. Much of the work had to be done remotely, which, while lowering radiation exposure, made it a very slow operation. In all, removing the 75 kg of sodium and bringing the cabin back to normal conditions took approximately three months.[12]

The *FBTR* has also seen several other accidents and unusual occurrences, such as unexplained reactivity transients.[13] Overall, the reactor's performance has been mediocre: it took fifteen years before the *FBTR* even managed fifty plus days of continuous operation at full power.[14] In the first twenty years of its life, the reactor has operated for only 36,000 hours, i.e. an availability factor of approximately 20 percent.[15] Despite this checkered history, IGCAR claims to have "successfully demonstrated the design, construction and operation" of a fast breeder reactor.[16]

The Prototype Fast Breeder Reactor

Even before the *FBTR* came on line, the DAE started making plans for a larger *Prototype Fast Breeder Reactor* (*PFBR*). In 1983, the DAE requested budgetary support from the Government.[17] The first expenditures on the *PFBR* started in 1987-88.[18] In 1990, it was reported that the Government had "recently approved the reactor's preliminary design and has awarded construction permits" and that the reactor would be on line by 2000.[19] Construction of the reactor finally began in October 2004 and was projected to be commissioned in 2010.[20] The *PFBR* will likely suffer from the two problems that have plagued breeder reactors elsewhere: the risk of a catastrophic accident and poor economics. It will also be a source of weapon-grade plutonium that might be used for the strategic program. See the discussion in chapter 2 of France's use of its first demonstration breeder reactor *Phénix* to produce weapon-grade plutonium for France's weapons program.

Safety

There are a number of reasons to doubt the safety of the *PFBR* design.[21] As with other breeder reactors, the *PFBR* design is susceptible to catastrophic accidents involving large and explosive energy releases and dispersal of radioactivity following a core meltdown. The potential for such Core Disruptive Accidents (CDA) comes from the reactor core not being in its most reactive configuration. If conditions during an accident cause the fuel bundles to melt and rearrange, the reactivity could increase leading to further core rearrangement and a potential positive feedback loop. Another unsafe feedback effect that is present in the *PFBR* design is its positive sodium void coefficient. This means that if the coolant heats up and becomes less dense, forms bubbles, or is expelled from the core, the reactivity increases. The magnitude of the void coefficient is a measure of the feedback and tends to increase with core size.[22] For the core design that has been adopted for the *PFBR*, it has a value of $4.3.[23]

Compounding the safety risks that come with this large and positive sodium void coefficient, the *PFBR* design also has a relatively weak containment, which is designed to withstand only 25 kilopascals (kPa) or one quarter of an atmosphere of overpressure.[24] This maximum overpressure that the *PFBR* containment is designed for is lower than some other demonstration reactors (table 3.1). If one considers the ratio of the containment volume times its design overpressure divided by the reactor power, V*P/E, the *PFBR* containment is weaker than those of all other demonstration breeder reactors except the *Prototype Fast Reactor* (*PFR*).[25] The difference appears more acute when the higher positive sodium void coefficient of the *PFBR* in comparison to other breeder reactors is taken into account

It is of course possible to design containments to withstand much higher pressures. Containments for light-water reactors routinely have design pressures above 200 kPa.[26] The DAE justifies this choice of containment design by arguing that its safety studies demonstrate that the maximum overpressure expected in a CDA involving the *PFBR* is smaller than this overpressure. But these results are based on favourable assumptions, in particular, that only limited parts of the

reactor core would participate in the CDA and that approximately 1 percent of the thermal energy released would be converted into mechanical energy. Based on such assumptions, the DAE estimates that the maximum credible energy release in a CDA is 100 megajoules (MJ).[27] It then calculates that such a CDA leading to sodium leakage into the containment would result in a containment overpressure of 20 kPa.

Name	Thermal Power E (MWt)	Sodium void coefficient ($)	Containment Volume V (m³)	Design overpressure P (kPa)	V*P/E (kNm/MWt)
Phénix	563	-	31,000	40	2200
PFR	650	2.6	74,000	5	569
CRBRP	975	2.29	170,000	170	29,600
SNR-300	762	2.9	323,000	24	10,200
MONJU	714	-	130,000	30	5460
PFBR	1250	4.3	87,000	25	1740

Table 3.1 **Containment design specifications of demonstration fast reactors.**
Source: Calculations based on data from IAEA, Fast Reactor Database 2006 Update.

There are, however, good reasons to consider much larger energy releases from a worst-case CDA to the extent of several hundreds of megajoules in the evaluation of the safety of a reactor design, especially one as large as the *PFBR*. Table 3.2 shows that the calculated CDA energy releases for a number of breeder reactors are much higher than that of the *PFBR*, both absolutely and when scaled by reactor power.

The energy releases from core collapse depend sensitively on the reactivity insertion rate, which is the rate at which the fuel rearrangement increases (inserts) reactivity.[28] The DAE's calculation of the maximum CDA energy release is based on a reactivity insertion rate of $65/s, which itself is the result of assuming only limited core disassembly.[29] There is ample reason and precedent to use an insertion rate of $100/s as a benchmark for disassembly calculations, with the caveat that it still is not quite an upper bound.[30] Likewise, the efficiency of conversion could be much larger than the 1 percent assumed by the DAE. Tests at the UK's Winfrith facility with core melt amounts of up to 25 kg suggest energy-conversion efficiencies of approximately 4 percent.[31] For a reactivity insertion rate of 100 $/s, and an energy conversion efficiency of 1 percent, the energy release from a CDA is 650 MJ.[32] It has been estimated that a 650 MJ CDA could lead to an overpressure of approximately 40 kPa on the containment, clearly much higher

than the design limit of the containment building.[33] Higher conversion factors would imply higher mechanical energy releases and thus higher overpressures and higher likelihoods of containment failure.

To summarize, there are good reasons to believe that the containment of the *PFBR* does not offer adequate protection against a severe CDA, especially given the many uncertainties inherent in calculations of CDA release energies.

Reactor	Year Critical	Power (MWt)	Approximate Maximum CDA Work Energy (MJ)	CDA/Power Ratio
Fermi	1963	200	2000	10
EBR-II	1964	65	600	9.2
SEFOR	1969	20	100	5
PFR	1974	600	600-1000	1-1.7
FFTF	1980	400	150-350	0.4-0.9
SNR-300	1983 (anticipated)	760	150-370	0.2-0.5
PFBR	2010	1200	100	0.083

Table 3.2   **Maximum CDA work energy calculations for fast breeder reactor systems.**

Economics

The main argument offered for the DAE's pursuit of breeder reactors is that India has only "modest uranium reserves" of approximately 60,000 tons, "which can support 10 000 MWe (megawatt electric) of *PHWR* (pressurized heavy-water reactor) capacities".[34] While widely repeated, this formulation is misleading. India's uranium resource base cannot be represented by a single number. As with any other mineral, at higher prices it becomes economic to mine lower grade and less accessible ores. Exploiting these would increase the amount of uranium available. Therefore, uranium resources can only be specified as a function of price.

As a way of evaluating the economics of breeder reactors, the cost of generating electricity at the 500 MWe *PFBR* can be compared with that at the *PHWR*,[35] the mainstay technology of the Indian nuclear program.[36] In order to address the argument about India's limited uranium reserves, it must be understood that the reserves are a function of uranium price, which allows calculation of the crossover price when the two technologies generate electricity at the same cost.

The total construction cost of the *PFBR* is estimated as Rs. 34.92 billion (mixed year Rupees; overnight construction cost of $646 millions in 2004 dollars). The overnight unit cost is $1292/kilowatt (KW) and is lower than the corresponding figure for recent Indian *PHWR*s of $1371/KW. This is quite in contrast to experiences around the world that suggest that breeder reactors are much more expensive than water moderated reactors; for light-water reactors, a typical estimate of the minimum cost difference is $200/kilowatt electric (KWe).[37] The *PFBR*'s estimated construction cost is also much lower than estimates of breeder reactor construction costs elsewhere; the Nuclear Energy Agency (NEA) gives a range of $1850-2600/KWe ($2000) or $2000-2800 (2004 dollars) for mixed-oxide fueled (MOX) fast reactors.[38] Actually constructed breeder reactors in other parts of the world also bear out the expectation of higher costs. Construction costs for the French *Phénix* reactor with a capacity of 250 MWe totalled $800 million in 1974 French FRF ($800 million in 2004 dollars) or $3200/KW. However, a further €600 million ($870 million in 2004 dollars) were spent on *Phénix* upgrades between 1997 and 2003. The 1240 MWe *Superphénix* was even more expensive. For these technical reasons, and the DAE's history of cost overruns at all the reactors it has constructed, it is likely that the *PFBR* capital cost will be higher than this projected value.[39]

In economic terms, the primary material requirement for the *PFBR* is plutonium. The *PFBR* design requires an initial inventory of 1.9 tons of plutonium in its core.[40] Based on a detailed model of the reactor, it has been estimated that at a 75 percent capacity factor, the *PFBR* requires 1012 kg of plutonium every year for refueling during equilibrium conditions.[41] The plutonium for the initial core and the first few reloads will have to come from reprocessing of *PHWR* spent fuel. At a real discount rate of 6 percent, reprocessing costs approximately $659 per kg of uranium in the fuel, which corresponds to a plutonium cost of $178/g.[42] Because of the higher plutonium content of the *PFBR* spent fuel, the unit cost of subsequent plutonium requirements would be lower, approximately $43/g.[43]

Following the Nuclear Energy Agency, the costs of fabricating breeder reactor core fuel and (radial) blanket uranium fuel have been assumed to be $1512/kg and $540/kg.[44] The base case assumes costs of $200/kg for natural uranium and $200/kgU for fabrication of uranium fuel for heavy-water reactors. The high base costs of uranium reflects the higher mining costs of poor quality uranium ore in India

Table 3.3 shows the difference in the levelised cost, at a real discount rate of 6 percent, of producing electricity at the *PFBR* and at the proposed 2 x 700 MW twin unit *PHWR*s.[45] The economics of the *PFBR* will be key to the future of breeder reactors in India. The DAE has argued that the "primary objective of the *PFBR* is to demonstrate techno-economic viability of fast breeder reactors on an industrial scale".[46] The results presented here show that the *PFBR* will not be viable, even at the projected costs and for optimistic assumptions about capacity factors. As table 3.4 shows, breeder reactors across the world have operated with relatively low cumulative load factors. There is no reason to expect that the *PFBR* experience

	PFBR (500 MWe)	PHWR (2 x 700 MWe)
Overnight Construction Cost (Million 2004 $)	646	1588
Real Discount Rate (%)	6	6
Capital Cost (Present Value) (Million 2004 $)	504	987
Capacity Factors (%)	80	80
Lifetime Plutonium/Uranium Cost (Million 2004 $)	1480	697
Total Lifecycle cost (Million 2004 $)	2212	2550
Levelised Cost (Rs/kWh)	2.77	1.54
Levelised Cost (cents/kWh)	6.30	3.49
Percentage Difference (PFBR-PHWR)	80 %	

Table 3.3 **Cost of electricity from breeder and heavy water reactors.**
All figures in 2004 U.S. dollars unless noted otherwise.

	PFR	BN-600	Phénix	Superphénix
Date of Construction Start	01-Jan-66	01-Jan-69	01-Nov-68	13-Dec-76
Date of First Criticality	01-Mar-74	26-Feb-80	31-Aug-73	07-Sep-85
Date of Grid Connection	10-Jan-75	08-Apr-80	13-Dec-73	14-Jan-86
Cumulative Load Factor	20.57%	71.51%	33.72%	6.6%

Table 3.4 **Reliability of breeder reactors.** Source: IAEA, PRIS Database.

would not be similar, and a capacity factor of 50 percent might well be more plausible. This would result in a levelised cost of 8.35 cents/kilowatt hours (kWh), 139 percent more expensive than *PHWRs*.

As mentioned earlier, the main rationale offered for the pursuit of expensive breeders is the shortage of uranium. The validity of this rationale has been examined by increasing the price of uranium from $200/kg to the crossover value where breeders become competitive. For the optimistic base case, with a *PFBR*

capacity factor of 80 percent, the levelised costs of electricity from the *PFBR* and *PHWR* are equal at a uranium price of $1375/kg. At a *PFBR* capacity factor of 50 percent, the crossover price is $2235/kg

These prices are much higher than current values and significantly larger quantities of uranium will be available at these prices. The distribution of uranium among the major geological reservoirs in the earth's crust corresponds to a roughly three hundred fold increase in the estimated amount of recoverable uranium for every ten fold decrease in the ore grade.[47] Based on this, and assuming that mining cost is inversely proportional to ore grade, one can surmise that the available uranium at costs less than $1375/kg and $2235/kg are approximately 124 and 417 times current reserves respectively. This is an underestimate because it ignores the general trends of reduced mining costs due to learning and improved technology.[48] In any case, India should have sufficient uranium for a nuclear energy sector based on PWHRs for many decades, with no reprocessing and breeder reactors.

Plutonium for weapons?
There may be another reason for the DAE's attraction to breeder reactors. This stems from the source of DAE's institutional clout: its unique ability to offer both electricity for development and nuclear weapons for security. This was revealed quite clearly during the course of negotiations over the U.S.-India nuclear deal, where in an ostensibly civilian agreement, much of the DAE's efforts were aimed at optimizing its ability to make fissile material for the nuclear arsenal within various constraints, especially the shortage of domestic low-cost uranium.[49] Most prominently, the DAE focused a lot of attention on keeping the fast breeder program outside of safeguards. In a prominent interview to a national newspaper, the head of the DAE said:

> Both, from the point of view of maintaining long-term energy security and for maintaining the minimum credible deterrent, the fast breeder programme just cannot be put on the civilian list. This would amount to getting shackled and India certainly cannot compromise one (security) for the other.[50]

In parallel, the DAE did not classify its reprocessing plants or its stockpile of reactor-grade plutonium as civilian. This allows for the possibility that breeder reactors like the *PFBR* could be used as a way to launder unsafeguarded reactor-grade plutonium, both in the historical stockpile as well as from future production at unsafeguarded reprocessing plants, into weapon-grade plutonium. While reactor-grade plutonium is consumed in the core of the *PFBR*, weapon-grade plutonium is produced in the radial and axial blankets. Based on neutronics calculations for a detailed three-dimensional model of the reactor, it has been estimated that 92.4 kg and 52 kg of weapon-grade plutonium will be generated in the radial and axial blankets (93.7 percent and 96.5 percent plutonium-239) respectively in the *PFBR* each year at a 75 percent capacity factor.[51]

If the blanket fuel elements are reprocessed separately rather than jointly with the core fuel elements, then the plutonium contained in them can be used for weapons. To make up for this, approximately 346 kg of reactor-grade plutonium derived from reprocessing spent fuel from India's *PHWR*s would have to be used in the *PFBR* annually. The existing stockpile of reactor-grade plutonium and *PHWR* spent fuel is adequate to meet this need for decades. Such a strategy would increase the DAE's weapon-grade fissile material production capacity several-fold.

Future projections
The *PFBR* is to be the first of the many breeder reactors that the DAE envisions building. The DAE's current projections are that nuclear power would grow to 20 gigawatt electric (GWe) by 2020 and to 275 GWe by 2052, including 260 GWe in metallic fueled breeders.[52] More recent media statements following the nuclear suppliers group lifting of its ban on nuclear trade with India project even larger rates of growth of India's breeder capacity. These projections seem to assume that spent fuel from imported light-water reactors fueled with imported uranium will be reprocessed and the plutonium extracted will also be used to provide startup fuel for breeder reactors.

These projections are primarily based on assumptions about the doubling time, the time it would take a breeder reactor to produce enough plutonium to fuel a new breeder reactor core. Since MOX fueled reactors have lower breeding ratios, by 2020 the DAE plans to switch to constructing breeders that use metallic fuel, which could have a much higher breeding ratio.[53] A higher breeder ratio will result in a shorter doubling time. The rate of growth also depends sensitively on the out-of-pile time, the time period taken for the spent fuel to be cooled, reprocessed, and fabricated into fresh fuel. The DAE optimistically assumes that all of this can be accomplished within one year.[54]

The DAE's methodology is flawed, however, and does not account correctly for plutonium flows.[55] To start with, the base capacity of metallic fueled breeder reactors (MFBRs) assumed for 2022 of 6 GWe, which is necessary for the 2052 projection, would require approximately 22 tons of fissile plutonium for startup fuel. The DAE does not have enough reprocessing capacity currently to handle all the spent fuel produced by the heavy water reactors that are operating and under construction. Even if the DAE does manage to inexplicably obtain the necessary plutonium to construct a MFBR capacity of 6 GWe with some to spare, under the DAE's assumed rate of growth, the plutonium stockpile would decline by approximately 40 tons just in the first ten years, even with an optimistic one year out of pile time. This is due to a three year lag between the time a certain amount of plutonium is committed to a breeder reactor and additional plutonium, which could be used as startup fuel for a new breeder reactor, is produced by reprocessing the irradiated spent fuel containing the initial plutonium.

A more careful calculation that takes into account the plutonium flow constraints shows that the capacity for MFBRs based on plutonium from the DAE's heavy water reactor fleet will drop from the projected 199 GWe to 78 GWe by 2052.[56] If the out-of-pile time were projected to be a more realistic three years, the MFBR capacity in 2052 based on plutonium from *PHWR*s will drop to 34 GWe.

While these figures may seem large compared to India's current nuclear capacity of only 4.1 GWe, they should be viewed in relation to the projected requirements, under business-as-usual conditions, of approximately 1300 GWe total generating capacity by mid-century. Further, the only constraint assumed here is fissile material availability. It assumes that there will be no delays due to infrastructure and manufacturing problems, economic disincentives due to the high cost of breeder electricity, or accidents. All of these are realistic constraints and render even the lower end of the 2052 projections quite unrealistic.

Conclusion

Breeder reactors have always underpinned the DAE's claims about generating large quantities of cheap electricity necessary for development. Today, more than five decades after those plans were announced, that promise is yet to be fulfilled. As elsewhere, breeder reactors are likely to be unsafe and costly, and their contribution to overall electricity generation will be modest at best.

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4 Japan's Plutonium Breeder Reactor and its Fuel Cycle

Tatsujiro Suzuki

This paper reviews the history, status, and probable future of fast reactor and associated fuel cycle development in Japan. The fast breeder reactor and its closed fuel cycle have been the cornerstone of Japan's nuclear-energy development program since the 1950s. For economic, technological, and political reasons, Japan's development and implementation of these technologies are significantly delayed. The budget for fast breeder reactor development has steadily declined since the mid-1990s, and its commercialization target has slipped from the 1980s to the 2050s. An accident at the *Monju* prototype reactor contributed to these delays and triggered a fundamental shift from research and development (R&D) and early commercialization to an emphasis on advanced fuel cycles.

Nevertheless, Japan is still committed to fast-reactor development. This paper examines the motivation for its continued commitment to a fast reactor program and concludes that several non-technological factors, such as bureaucratic inertia, commitments to local communities, and an absence of R&D oversight have contributed to this entrenched position. Japan is currently reorganizing its R&D programs with the goal of operating a demonstration breeder reactor by about 2025. This effort is in response to the government sponsored Nuclear Power National Plan and the Bush Administration's Global Nuclear Energy Partnership Program (GNEP). Breeder R&D programs face significant obstacles, such as plutonium-stockpile management, spent fuel management, fuel cycle technologies, and arrangements for cost and risk sharing between industry, national and local governments. As a result, it is likely that fast breeder reactor programs will continue to slip.

Program overview—history and status

Japan's fast breeder reactor program was conceived in the Japan Atomic Energy Commission's (JAEC) first Long Term Plan, published in 1956.[1] Among various reactor types under review, the JAEC selected the fast breeder reactor and its closed fuel cycle as the preferred technologies for R&D and endorsed the importation of light-water reactor technology from the United States.[2]

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The JAEC's 1967 Long Term Plan concluded that the fast breeder reactor should be the mainstream of future nuclear power generation[3] and the Government established the Power Reactor and Nuclear Fuel Development Corporation (PNC) as the primary R&D institution for fast breeder reactor and nuclear fuel cycle development. The Plan envisioned that an experimental fast reactor would be built during the 1970s, and the first commercial fast breeder reactor by the late 1980s.

Japan's first fast breeder reactor was the experimental *Joyo* (Eternal Sun), built at the Japan Nuclear Cycle Development Institute's Oarai Engineering Center. *Joyo* achieved criticality in 1977 at an initial power level of 50 megawatt thermal (MWt). Power was increased to 75 MWt in 1979, and to 100 MWt with its Mark II core, which achieved criticality in 1982. From 1983 to 2000, *Joyo* operated as an irradiation test bed for fuels and materials for future Japanese fast reactors. Since 2003, *Joyo* has been operating at 140 MWt with its Mark III core, and in April 2007 it completed its 6th duty cycle. By 12 March 2007, *Joyo* had operated for 70,000 hours. Thus, in the 30 years between 1977 and 2007 *Joyo* operated approximately 27 percent of the time.

The prototype fast breeder reactor *Monju* (280 megawatt electric) was developed in parallel with *Joyo*, but construction was delayed and it did not achieve criticality until 1994. On 8 December 1995, *Monju* experienced a serious sodium leak and fire when intense vibrations caused the failure of a thermocouple attached to the secondary sodium loop. The sodium reacted with oxygen producing a fire that melted the steel structures in the room. No injuries were reported and no release of radioactivity occurred since the sodium in the secondary loop was not radioactive.

PNC's cover-up of the accident caused a social and political uproar that delayed the repair and restart of *Monju*. In June 2001, PNC submitted a re-license application for *Monju*, which was granted in December 2002. Legal challenges against PNC surrounded the relicensing causing further delays and on 27 January 2003, the Kanazawa branch of Nagoya's High Court reversed its 1983 approval to build the reactor. Just over two years later, on 30 May 2005, Japan's Supreme Court ruled for PNC, thereby clearing all legal barriers for the restart of *Monju*. Restart was scheduled for October 2008 but as of January 2010 the reactor is still off-line.

Japan Atomic Power Company (JAPCO) finalized plans for a 660 megawatt electric (MWe) demonstration commercial fast breeder reactor in 1994. The project experienced delays because of the *Monju* accident and was eventually canceled in the late 1990s.

R&D on reprocessing fast reactor spent fuel started in mid-1970s, and reprocessing of *Joyo* spent fuel was conducted at the experimental Chemical Processing Facility (CPF) starting in 1982. Following the experience gained at the CPF, PNC started construction of a Recycle Equipment Test Facility (RETF) in 1995, which is the

first pilot-scale reprocessing facility for fast reactor spent fuel, the counterpart of the Tokai pilot reprocessing plant for light-water reactor spent fuel. The Tokai plant adopted imported French technology but the RETF intends to employ technologies currently under development under the cooperative program with Oak Ridge National Laboratory (ORNL) in the United States. The first phase of construction was completed in 2000, but its scheduled completion date is currently unknown.

Declining budgets and slipping targets

While the public commitment of Japan's Government to the fast breeder reactor and closed fuel cycle has not wavered, the fast breeder reactor R&D budget has steadily declined, and, by 1996 had dipped below a 10 percent share of the nuclear R&D budget. The fast breeder reactor program share of total nuclear R&D peaked at 35 percent in early 1970s during the construction of *Joyo*. In 1989 it fell to 20 percent (¥77 billion) during peak construction at *Monju*. Since 1989, both the fast breeder reactor budget and its share of Japan's total nuclear R&D budget have steadily declined. Cumulative spending on fast breeder reactor R&D from 1956 to 2007 was ¥1,480 billion, representing approximately 12 percent of total spending. Figure 4.1 shows the budget trends for all nuclear energy and fast breeder reactor R&D. The
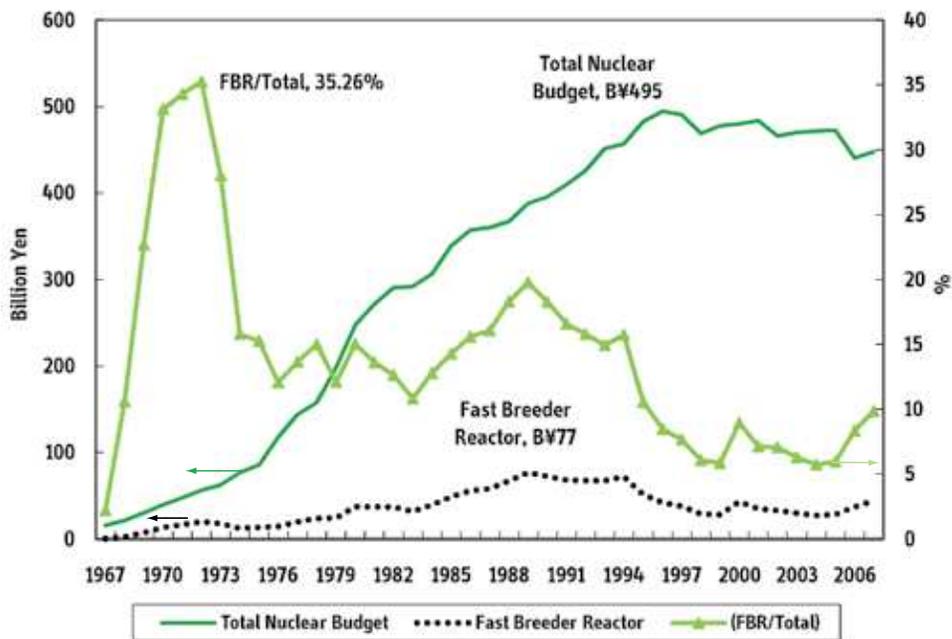

Figure 4.1 **History of Japan's R&D budgets for nuclear power and breeder reactors.**
Peak-year budgets and fast breeder reactor budget percentages are indicated.

target date for fast breeder reactor commercialization has slipped by 80 years in a period of 50 years. In 1956, the Long Term Plan anticipated commercialization in the 1970s. In 1967, the year that the PNC was established, fast breeder reactor commercialization was pushed back to the 1980s and the PNC decided that an Advanced Thermal Reactor (ATR) was required as an interim reactor between the light-water reactor and the fast breeder reactor. In 1987, the JAEC confirmed that light-water reactors would remain the main power generation source for the foreseeable future, and the commercialization target for fast breeder reactors was pushed back to the 2020–2030s. The most recent JAEC Framework for Nuclear Policy, which supersedes the Long Term Plan, has revised the goal for fast breeder reactor commercialization to approximately 2050 (table 4.1).[4]

Plan Year	Anticipated Completion	Comments
1956	1970	As a main source of power
1967	1980	An advanced thermal reactor is required as an interim solution
1987	~2020–2030	The light-water reactor is selected as the main source of power for the foreseeable future
2000	~2030 or later	Breeder reactors may be one of the future options
2006	2050 or later	

Table 4.1   **History of the commercialization schedule for breeder reactors in Japan.**

Priority shifts after the Monju accident

The *Monju* accident triggered a significant shift in Japan's fast breeder reactor program. After the accident, the JAEC established an ad-hoc "Roundtable Committee on FBR" to develop new policies. Prof. J. Nishizawa of Tohoku University, who was not a fast breeder reactor expert, chaired the committee. The Committee also included experts from outside the nuclear community, including Mr. Yukio Okamoto (ex. Ministry of Foreign Affairs), Prof. Sawako Takeuchi, an economist, and Prof. Hitoshi Yoshioka of Kyushu University, a nuclear critic. Although the Committee confirmed the continuation of fast breeder reactor development, it recommended a more realistic and flexible approach, declaring that the fast breeder reactor should be considered as a promising option (rather than the ultimate goal) and suggested "periodic review of R&D programs from the standpoint of technological and economic feasibility."[5] It also endorsed a more diversified R&D program to explore technical alternatives to existing fast breeder reactor technologies.

Following this report, the JAEC's Long Term Plan, published in 2000, established a goal "to maintain the technological option of the fast breeder reactor and its associated fuel cycle...in order to prepare for future energy problems," and recommended programs to explore "various alternatives to currently developed sodium-type fast breeder reactor and PUREX (wet) reprocessing technology."[6]

Status

The 2005 Long Term Plan was renamed the Framework for Nuclear Energy Policy and established a new fast breeder reactor commercialization target of 2050.[7] In 2006, the Sub-committee on Nuclear Energy Policy of the Government's Advisory Council on Energy published Japan's Nuclear Power National Plan, which laid out detailed policy measures based on the JAEC's framework.[8] The Nuclear Power National Plan reiterates the 2050 commercialization target for the fast breeder reactor and announced a goal of developing a post-*Monju* demonstration fast breeder reactor by 2025. The associated Phase II "Feasibility Study on Commercialization of Fast Reactor Cycle Systems" compared various types of fast reactor designs and associated fuel cycle technologies, and tentatively identified a sodium-cooled fast reactor with advanced wet reprocessing technology as the preferred option.[9]

The study compared four fast-neutron reactor designs: sodium-cooled (1.5 gigawatt electric (GWe)) with metallic fuel, helium-cooled (1.5 GWe) with nitride fuel, lead-bismuth-cooled (0.75 GWe) with nitride fuel, and water-cooled (1.356 GWe) with mixed oxide fuel (MOX). Unit construction cost estimates for a sodium-cooled fast breeder reactor would be the lowest ¥180,000/kilowatt compared with approximately ¥200,000/kilowatt electric for the other designs. Four basic options for advanced reprocessing and fuel technologies were evaluated:

1. Advanced wet reprocessing plus simplified pelletized MOX fuel;
2. Metal electro-refining reprocessing plus injection cast metallic fuel;
3. Advanced wet reprocessing plus vibration packing (Sphere-pack) MOX fuel; and,
4. Oxide electro-refining reprocessing plus vibration packing (Vipac) MOX fuel.

The most economical option would be the advanced wet reprocessing plus simplified pelletized MOX fuel in a large (200 ton/year) plant (~¥0.5–0.66/kWh) with the alternatives costing up to ¥1.6/kWh. None of these cost estimates are engineering estimates. All represent development targets required for fast breeder reactors to be competitive with light-water reactors.

The Nuclear Power National Plan also set out important principles for the future development of fast breeder reactor and fuel cycle systems. First, it established a cost-sharing principle to distribute demonstration fast breeder reactor project

costs between the utility companies and the Government. It specified that the private sector would invest an amount equivalent to the cost of a commercial light-water reactor, significantly reducing the financial risk for utilities.[10]

Another important principle of the Nuclear Power National Plan was that the second commercial reprocessing plant after the Rokkasho plant should be timed to match the pace of fast breeder reactor development and deployment. It suggested that planning for the second reprocessing plant start around 2010.

In 2007, the Government increased the fast breeder reactor R&D budget for the first time since the late 1990s to ¥44 billion in response to these new programs and principles. It is now approximately 10 percent of the total nuclear budget. This budget increase was prompted partially by international developments, notably the announcement of GNEP, which had an initial emphasis on using fast-neutron reactors to fission plutonium and other transuranic elements in light-water reactor spent fuel.

The socio-political factors behind Japan's entrenched commitment to fast breeder reactor technology

Despite the marked slippage of fast breeder reactor commercialization targets, why have Japanese commitments to the fast breeder reactor remained, at least publicly, unchanged? There are three possible explanations.

Organizational Commitments

In 1967, a special law established PNC with the mission to develop indigenous fast breeder reactors and their associated fuel cycle technologies. This mission endured after the *Monju* accident in 1995 when PNC was renamed the Japan Nuclear Fuel Cycle Development Institute (JNC). JNC subsequently merged with the Japan Atomic Energy Research Institute (JAERI), a national research institution responsible for fundamental nuclear technology (including fusion) and nuclear safety research and in 2006 it became the Japan Atomic Energy Agency (JAEA). JAEA was established with the continued mission of developing fast breeder reactor and fuel cycle technologies. With this legal commitment to fast breeder reactor programs, it may not be easy for Japan to change its nuclear research agenda.

Local Politics

Local politics with respect to nuclear facilities is complex and influential. Government financial incentives, called kofu-kin, reward communities for accepting nuclear-related facilities and play a large role in local politics. Once a local community accepts a nuclear facility, it receives annual payments (in billions of yen) from the Government. Kofu-kin and tax revenues from nuclear facilities become a major component of local budgets. Therefore, despite strong resentment about the cover-up after the *Monju* accident, the local community has a significant incentive for restarting the plant.

Another factor driving fast breeder reactor and fuel cycle policies is the difficulty of finding off-site spent fuel storage sites. Because on-site storage pools are reaching their capacity, reprocessing is seen by many as the only alternative. The rationale for reprocessing becomes more persuasive if it paves the way towards the commercialization of fast breeder reactors.

Lack of Oversight

JAEC is the primary government entity authorized to review and make decisions on Japan's nuclear R&D programs. While JAEC may advise R&D institutions to revise their goals and schedules, it typically endorses their R&D plans.

In 2001, the Council for Science and Technology Policy (CSTP) was established by the Basic Law on Science and Technology within the reformed Prime Minister's Office and is chaired by the Prime Minister. Its primary function is to review R&D plans submitted by government agencies. It grades major R&D programs from S (most important) to A, B, C (least important). It is intended to strengthen the Prime Minister's ability to override agency R&D budgets driven by vested interests. The *Monju* project received a grade of "S" and the Feasibility Study on Commercialization of Fast Reactor Cycle Systems[11] (FaCT) program received an "A" and therefore it is unlikely that CSTP will override development plans for the *Monju* project or the FaCT program.

Future prospects and major issues

Although the Nuclear Power National Plan set a goal for completion of a demonstration fast breeder reactor by 2025 and commercialization by 2050, there are obstacles that may compromise these goals.

One obstacle is plutonium stockpile management. Japan has more than 46 tons (8.7 tons in Japan, approximately 37 tons in Europe) of separated plutonium in stock, but its MOX recycling program has made little progress. When the Rokkasho reprocessing plant (800 tons heavy metal/year capacity) begins full operation, the stockpile is likely to increase. Since reducing the plutonium stockpile should be a top priority for Japan, breeding is not likely to be an important policy goal for Japan's nuclear power program.

A second obstacle relates to spent fuel management and its impacts on fuel-cycle technology. Japan has been reviewing various reprocessing and MOX fuel fabrication methods, including pyro-processing technology developed in the United States for reprocessing fast reactor metallic fuel. Historically, spent fuel management, and not plutonium demand, has driven Japan's reprocessing requirements. If this focus is maintained, it is likely that Japan will build a second plant, using wet technology, to reprocess uranium oxide spent fuel. So far, Japan's R&D on reprocessing technologies has focused on the classic PUREX process.

If Japan pursues its MOX-recycling plans, spent MOX fuel will accumulate and Japan may want to reprocess this fuel. The technological choice for the second reprocessing plant is a complex policy issue.

A third obstacle is the matter of cost and risk sharing among stakeholders. Overall, it is not clear how much fast breeder reactor fuel cycle programs will cost and who will bear those costs. The Nuclear Power National plan proposes a cost sharing arrangement for a demonstration fast breeder reactor, but future cost sharing arrangements are uncertain. Meanwhile, one of the goals set by the Ministry of Economy, Technology and Industry's next generation light-water reactor program is to extend the life-times of the reactors to 60-80 years. If this goal is achieved, the need for the fast breeder reactors may not materialize even after 2050.

Conclusion

Japan remains officially committed to the fast breeder reactor and closed fuel cycle systems. However, the fast breeder reactor commercialization date has receded far into the future while the fast breeder reactor R&D budget has been shrinking. Japan's continued commitment to the fast breeder reactor appears largely driven by socio-political factors affecting Japan's management of the back-end of the light-water reactor fuel cycle and R&D management. The Nuclear Power National Plan restated Japan's interests in fast breeder reactor and advanced fuel cycle programs due in part to international developments, especially the GNEP initiative, which has since lost support in the Obama Administration and in the U.S. Congress.

Endnotes

1   Japan Atomic Energy Commission, "Long Term Plan for Research, Development and Utilization of Nuclear Energy," 1956. (in Japanese)

2   In a "closed" fuel cycle a chain-reacting material, such as plutonium, is separated out of spent fuel and recycled.

3   Japan Atomic Energy Commission, "Long Term Plan for Research, Development and Utilization of Nuclear Energy," 1967. (in Japanese)

4   "Framework for Nuclear Energy Policy," October 11, 2005, <http://www.aec. go.jp/jicst/NC/tyoki/taikou/kettei/eng_ver.pdf> (accessed May 2009).

5   Report by the Japan Atomic Energy Commission Roundtable Committee on Fast breeder Reactor, "Basic Policy towards development of FBR," December 1, 1997.

6   Japan Atomic Energy Commission, "Long Term Plan for Research, Development, and Utilization of Nuclear Energy," 2000.

7   Japan Atomic Energy Commission, "Framework for Nuclear Energy Policy," October 11, 2005 <http://www.aec.go.jp/jicst/NC/tyoki/taikou/kettei/eng_ver. pdf> (accessed May 2009).

8   Ministry of Economy, Trade and Industry, Advisory Committee on Energy, Committee on Electricity Industry, Sub-committee on Nuclear Energy Policy, "Nuclear Power National Plan," August 2006, <http://www.meti.go.jp/report/ downloadfiles/g60823a01j.pdf > (accessed May 2009, in Japanese).

9   Wet reprocessing refers to variants of the nitric-acid-based PUREX process originally developed by the United States to recover plutonium for its nuclear-weapon program. Pyroprocessing would be an example of a dry, (waterless) process in which spent fuel is dissolved in molten-salt, the heavy-metal oxides reduced to metals by reacting them with lithium and the precipitated heavy metals electrorefined to separate the transuranics from the uranium.

10 This reflects the experience of *Monju*. *Monju*'s construction cost increased from ¥400 billion to ¥590 billion, the utilities' share was originally set to be approximately 15 percent of total cost (i.e. ¥60 billion). Utilities finally agreed to pay ¥109 billion, which was equivalent to the light-water reactor construction cost (per kilowatt). See Eugene Skolnikoff and Tatsujiro Suzuki, "International Responses to Japanese Plutonium Programs," MIT Center for International Studies Working Paper, #2614 C/95-5, August 1995, 4.

11 The FaCT program was conducted from 1997 to 2006, exploring various types of fast reactor and fuel cycle technologies in order to select commercially viable fast reactor technologies, <http://www.jaea.go.jp/04/fbr/top.html> (accessed May 2009).

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5 The USSR-Russia Fast-Neutron Reactor Program

Gennadi Pshakin

The Soviet Union's fast-neutron reactor program began at the end of 1949 when physicist Alexander Leypunsky presented a special report to the Government on the idea of creating nuclear reactors that could produce more fissile material than they consumed. The rationale offered was that in the future, as the Soviet nuclear industry expanded rapidly, there would be a shortage of uranium. In November 1949, the Government decided to launch a fast-reactor development program. Leypunsky was designated as the program's scientific leader and the State Scientific Centre of Russian Federation, Institute of Physics and Power Engineering (IPPE) in Obninsk became the lead research institute.[1]

The program contended with inadequate knowledge in many areas, including the behavior of the candidate reactor, core and coolant materials under irradiation and the information required to design the steam generators, where the reactor coolant and water would be separated only by a thin layer of material.[2]

It is important to note that the program started only four years after the most destructive war the country had ever faced. There were shortages of both special materials and personnel with relevant expertise.[3]

The first decade, 1949-59
The first decade of the Soviet breeder program was exploratory. In May 1955, a fast critical assembly *BR-1* (in Russian "*Bystry Reactor-1*," i.e. *Fast Reactor-1*) started operation at IPPE. It was fueled with metallic plutonium and without a coolant.[4] The compact plutonium core and uranium blanket allowed a breeding coefficient of approximately 1.8, which lent great support to the breeding idea.

The following year, the fast reactor, *BR-2* began operation. Both gaseous and liquid-metal coolants were considered during the design stage.[5] Mercury was chosen but the metal plutonium fuel was not stable under irradiation even at low temperatures and mercury leaked from pipe joints and corroded the steel cladding.[6]

The *BR-2* was replaced with the *BR-5* (5 MWt),[7] and commenced operations in 1959. It was cooled with liquid sodium and fueled with plutonium dioxide to allow higher fuel temperatures and power densities (up to 500 kilowatts/liter) in the core.[8] The *BR-5*'s power was subsequently increased to 10 megawatt thermal (MWt) and it operated until 2004. In addition to reactor research and development, the *BR-5* was used for medical-isotope production and even medical treatment (neutron-capture therapy of throat cancer using neutron beams from the reactor).

Second and third decades, 1960-80

During the second and third decades of the program, experience was acquired in the use of fast-reactor technology.

In 1961, the critical assembly *BFS–1* started operation at IPPE. It allowed researchers to simulate fast-reactor core volumes of up to 3 m^3 with cores fueled by different mixtures of plutonium and uranium of varying enrichments, and different configurations of control and safety rods. It also allowed studies of the effects of sodium voids on reactivity and other physical effects. *BFS-2*, which started operating at IPPE at the end of the 1960s, could simulate cores with volumes up to 10 m^3.

A higher power special fuel-testing reactor, the *BOR-60*, was designed and constructed in the Institute of Atomic Reactors (Dimitrovgrad) in five years and began operating in 1969. Vibro-packed fuel was tested in this reactor. It is still operational.

Between 1962-1964, the future direction of Soviet nuclear energy development was studied. A main concern was conservation of uranium resources. The study concluded that a "promising perspective is expansion of nuclear energy using fast breeder reactors starting with enriched uranium fuel and step-by-step replacement with plutonium fuel."

A demonstration project was initiated even before the *BR-5* began operating. Initially the demonstration reactor was named *BN-50* (50 MWt) but later the power was increased to 1000 MWt. The reactor came to be called *BN-350* for its equivalent electrical output.[9] The design of the demonstration *BN-350* and a significant number of experiments at the *BFS-1* critical assembly were completed before construction started in 1964.[10] The Minister of Atomic Energy, Yefim P. Slavsky, decided to build the reactor on the Mangyshlak peninsula on the Caspian Sea. The heat was used for desalination as well as electricity generation. It was fueled with uranium enriched up to 20-25 percent uranium-235 and with mixed-oxide uranium-plutonium (MOX) test fuel assemblies. It began operations in 1972.

A year later, in late 1973, the *BN-350* experienced a major sodium fire due to the failure of one of the steam generators. The *BN-350* steam generators were designed and built without sufficient experimental study. Additionally, welding quality control on the first set of steam generators was inadequate. The reactor was shut down for repair for approximately four months and then continued operations until it was shut down permanently in April 1999.

Even before the *BN-350* began operating, the Government decided to start a second fast-neutron reactor with a still higher power as a step toward fast-neutron reactor commercialization. The project was called *BN-600* (600 megawatt electric). Experience acquired during the initial period of *BN-350* operation was used to make changes to the *BN-600* design.

The reactor was designed with a secondary sodium circuit between the radioactive primary sodium and the steam generator. It is a pool-type design with the heat exchangers between the primary and secondary sodium loops within the reactor vessel. There is no containment structure. The reactor was the third unit of the Beloyarskaya nuclear power plant in the Ural region and is still operating.

As of 1997, there were 27 sodium leaks in the *BN-600*, 14 of which resulted in sodium fires. The largest leak was 1000 liters.[11] The fires were extinguished without casualties, however, and plant personnel repaired the damage. The steam generators are separated in modules so they can be repaired without shutting down the reactor.

No irresolvable problems were encountered during construction of the *BN-350* and *BN-600* reactors. The pumps, vessel, piping, cover of the reactor with its movable port for locating the refueling machine over a specific fuel assembly, and steam generators were produced at Soviet manufacturing plants, and all mechanical equipment was tested prior to final installation. Standard turbines were used.[12]

During 1970-80, IPPE launched the designs of two new fast-neutron reactors, the *BN-800* (figure 5.1) and *BN-1600*. The *BN-800* (800 MWe), which is again under construction (as of 2009), will be a modernized version of the *BN-600* to match a standard turbine.[13] The *BN-1600* will be a commercial nuclear power plant. In the early 1980s the Government planned to build five *BN-800*s in the Ural region. After the Chernobyl accident in 1986, however, the Soviet nuclear energy program was cut back (table 5.1).Russia's economy was not able to support substantial investments in new nuclear power plants during the 1990s. In addition, fast-neutron reactors were not economically competitive with Russia's light-water and graphite-moderated thermal-neutron reactors and estimates of available high-grade uranium increased sharply as a result of the discovery of large uranium deposits in Kazakhstan in the 1960s and 1970s.

	1966-1975	1971-1980	1981-1990
Planned nuclear capacity additions (GWe)	12	27	67
Realized nuclear capacity additions (GWe)	4	10	25

Table 5.1 **Planned and realized nuclear capacity additions in the Soviet Union.** Gigawatt electric (GWe).
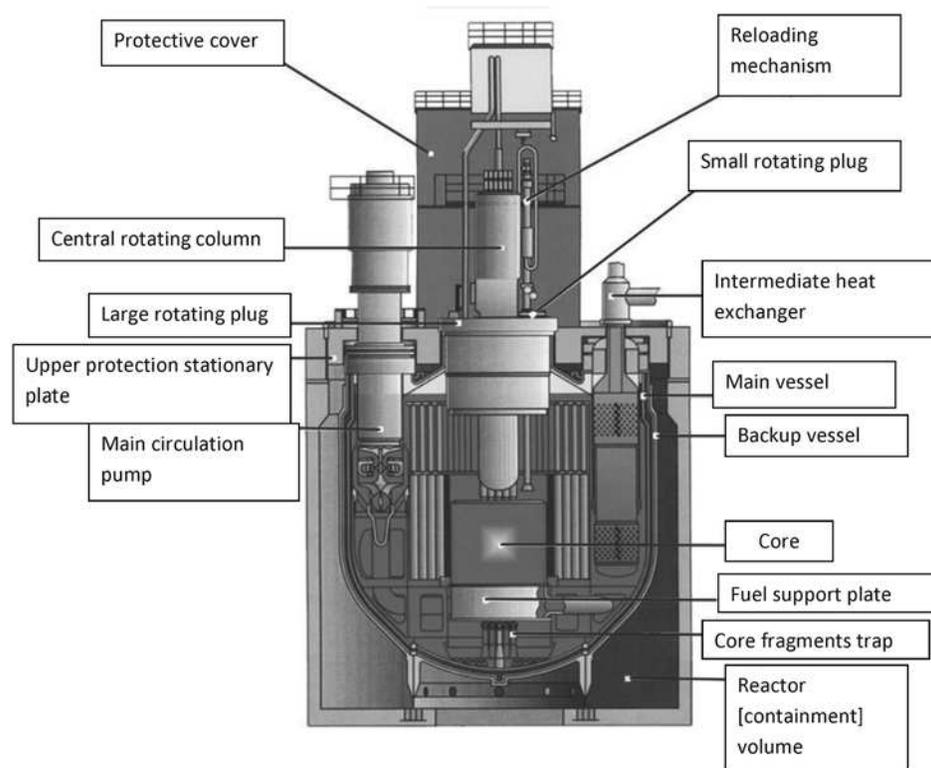

Figure 5.1 **Artist's Rendition of the BN−800 reactor now under construction (2009).** Source: Institute of Physics and Power Engineering.

In 2000, however, in a speech at the U.N. Millennium General Conference, President Putin unveiled a new program for expansion of Russia's nuclear capacity. This expansion program, while focused primarily on light-water reactors, includes fast-neutron reactors. The first step towards commercialization will be the construction of a few replicas of the *BN-800* and completion of the design of a commercial prototype of the *BN-1600*.[14]

The fast-neutron reactor program has several goals:

1. Develop a closed uranium-plutonium fuel cycle;

2. Produce chain-reacting uranium-233 from neutron capture in thorium blankets as a potential fuel for thermal-neutron reactors;

3. Fission the minor transuranics, neptunium, americium and curium; and,

4. Significantly reduce highly radioactive waste volume for a final geological repository.

It is difficult to estimate the cumulative investment in the fast-neutron reactor program. One estimate offered in 2004, by F.M. Mitenkov of the Afrikantov Experimental Machine Building Design Bureau, is approximately $12 billion, which included construction of the *BN-600* and design of the *BN-800*.[15]

Safety of fast-neutron reactors

From the very beginning, leaders of the fast-neutron reactor development program had safety concerns. Reactivity safety was studied theoretically using a number of criticality experiments under different scenarios such as refueling, transition from subcritical to supercritical, and the effectiveness and safety of the control rods. These findings were subsequently supported by practical experience with the *BOR-60*, *BN-350* and *BN-600*. Additionally, special experiments were performed to study fires resulting from sodium leaks into the air (which can be effectively suppressed) and into the water in steam generators.

A more significant problem centered on the construction of the steam generator. Two types of steam generators were tested, water in pipe surrounded by sodium (straight-type) and sodium in pipe surrounded by water (reverse-type). Experience acquired from both types revealed that the reverse-type is safer and led to the idea of including an intermediate sodium heat-transfer loop between the radioactive sodium primary coolant and the steam generator.

Economics

Because of the added secondary circuit, the total amount of structural material in the *BN-600* is approximately 50 percent more than for the *VVER-1000* (1000 MWe) light-water reactor.[16] The estimated cost of *BN-800* construction is $2.2-2.5 billion, approximately 11 percent greater than that of the standard *VVER-1000* or costing approximately 40 percent more per kilowatt (KW).[17] This higher capital

cost and the higher cost of plutonium fuel relative to low-enriched-uranium fuel would make electricity from the *BN-800* much more expensive than that from the *VVER-1000.*

Scientific problems addressed, solved and remaining

Most of the technical problems relating to fast-neutron reactors were solved through extensive experimental and theoretical studies performed during the first 40 years of the program. Various reaction cross sections were measured for neutron calculations (criticality, neutron flow distribution, reactivity effects, control-rod effectiveness, etc.). The results were replicated in a number of criticality experiments. The criticality of the *BN-350* was predicted within 1 percent (198 fuel assemblies calculated versus 200 experimental). Control-rod effectiveness was estimated with less than 10 percent uncertainty and temperature and power reactivity coefficients with 15-20 percent uncertainty. The startup measurements on the *BN-600* produced similar results. There has been significant progress towards the understanding of the swelling effects in steel from high neutron fluence ($>10^{22}$ n/cm^2).[18] The behavior of liquid metals, particularly liquid sodium, was studied at a number of IPPE's experimental facilities over 50 years. Practically all aspects were studied and the results explained theoretically. New requirements for nuclear safety promulgated after the Chernobyl accident will require additional study but will likely not raise new scientific obstacles.

Technological problems discovered, resolved and remaining

From the very beginning there were questions about how to remove heat produced in a fast-neutron reactor's compact core. Sodium was chosen as the best coolant based on theoretical research and experiments.

A serious drawback of sodium is that it burns in water. A number of experiments at IPPE and experience gained with the *BR-5*, *BN-350*, *BOR-60* and *BN-600* suggested that this problem is not a major issue for fast-neutron-reactor safety. Even the 1973 sodium fire at the *BN-350* did not affect reactor safety. Problems with steam-generator design were corrected step-by-step.

Fortunately, before the startup of construction of the *BN-350*, it was discovered that steel swelling under high neutron fluence was problematic. At the last moment, the fuel assembly design was modified to take into account the swelling and the planned burn-up of the fuel was limited (though subsequently increased on the basis of further experience). Irradiation tests on different types of steel were made in the *BOR-60*, *BN-350* and *BN-600* reactors for consideration in future projects.[19]

Mercury was initially considered as a coolant but is highly corrosive to most reactor materials. Although sodium-potassium alloy is a good coolant (with a low melting temperature of approximately 20 °C so that a heating system is not required for liquefaction), the alloy is more flammable then pure sodium.

Lead and bismuth and their alloys are more promising as fast-reactor coolants. Neutrons do not lose a significant amount of energy when they collide with the heavy nuclei of these elements. They are not flammable and do not react with water. At the same time, however, they are significantly more corrosive to steel than sodium with corrosion properties that are dependent on the oxygen content in the alloy. Research on lead-bismuth alloys was initiated in 1951.[20] They are effective coolants for compact nuclear reactors, which is why they were used for submarines.[21] During the past decade, the alloy was considered as a coolant for a new type of fast-neutron reactor, the BREST project. Rosatom[22] has decided to build an experimental 75 MWt reactor with lead-bismuth coolant (*SVBR-75/100*) before developing a commercial prototype.[23]

Thus far, most of the fuel in the *BN-350* and *BN-600* reactors has been uranium dioxide. Some experience with mixed-oxide (MOX) uranium-plutonium fuel was acquired in the *BOR-60* reactor and in a few experimental fuel assemblies in *BN-350* and *BN-600*. Limited experience in carbide and nitride fuel was gained with the *BR-5/10* but not enough to deploy these fuels in a future fast-neutron reactor.

Recently, the director general of the Research Institute of Atomic Reactors, Alexander V. Bychkov, declared that most of the problems of vibro-packed fuel-fabrication technology have been solved and that it is ready for commercial implementation.[24] However, full-scale experiments with closed fuel cycles have not been conducted.

Seven tons of *BOR-60* fuel have been reprocessed, 4 tons of which were MOX and some of the separated plutonium was recycled. A number of questions are unresolved. How will a transition between bench scale and commercial scale technology influence the quality of the fuel pins and assemblies? If pyro-chemical processing is used, the degree of separation of the fission products will have to be determined.

Endnotes

[1] O.D. Kazachkovski, *Fast Neutron Reactors. Science-technical Memoirs* (Obninsk: IATE, 1995); "State Scientific Centre of Russian Federation, Institute of Physics and Power Engineering–50 years" (Proceedings, Institute of Physics and Power Engineering, Obninsk, 1996).

[2] From personal conversation with the founding members, it was not possible to learn what Soviet scientists knew about either the secret Los Alamos report by E. Fermi, "About Breeding," dated April 1944, or about the first experimental U.S. fast-neutron reactor, *Clementine*, built in 1946.

[3] In 1945, the Atomic Project's security service found that the country had only 1500 physicists, including university teachers. Most of the specialists for the atomic project were transferred from other industries.

[4] The purchase of 12 kilograms of weapon-grade plutonium from the military program was a personal decision of the Minister E. Slavsky; it is worth noting that plutonium production for the military program had just started. Total power was 100 watts.

[5] Gases (helium, N_2O_4) and liquid metals (sodium, potassium, lithium, bismuth, lead and mercury). An experimental facility with a lead-bismuth (Pb-Bi) alloy was built and began operating in 1951. A sodium experimental facility was built in 1953. Eventually the Pb-Bi coolant was selected for submarine propulsion reactors.

[6] State Scientific Center of the Russian Federation – Institute for Physics and Power, "History of Peaceful Use of Atomic Energy at USSR," Documents and Materials (1994): 106.

[7] MWt denotes thermal megawatts; MWe denotes electrical megawatts.

[8] L. Kochetkov, "From Mercury to Sodium," interview published electronically by ATOMINFO.RU (in Russian), <http://www.atominfo.ru/news/air3253.htm>, (accessed 24 June 2009).

[9] It was risky to make this leap in scale but leaders of the program were eager to make rapid progress in fast-neutron reactor development. The main parameters of the reactor core were tested in the *BR-5*, however, including the specific power and the coolant outlet temperature. The reactor design was loop-type with 6 coolant loops including one in hot reserve.

[10] Design of the *BN-350*, and later fast-neutron reactors, was entrusted to the Afrikantov Experimental Machine Building Design Bureau in Nizhny Novgorod (Gorky in Soviet times). This organization had significant experience in designing and testing artillery systems and had great expertise in quality control. It also worked to maintain simplicity in operator controls and introduced "test-before-install" principles.

[11] O. M. Saraev, "Operating Experience with Beloyarsk Power Reactor BN600 NPP, Beloyarsk Sverdlovsk Nuclear Power Plant," IAEA FBR database, <http://www.iaea.org/inisnkm/nkm/aws/fnss/fulltext/1180_4.pdf> (accessed 24 June 2009).

12 The vessels for *BN-350* and especially for *BN-600* were produced in pieces that were welded together at the site. It is important to note that the low-pressure vessels used for fast-neutron reactors are much thinner than the high-pressure vessels used for light-water reactors. During construction of *BN-350* there was intensive cooperation between Czechoslovakia and the USSR on fast-neutron-reactor development. A few steam generators were produced at the Shkoda plant in Czechoslovakia and installed in the *BN-350.*

13 "Engineering of NPP Unit with Fast Neutron Reactor *BN-800*," *Atomic Strategy 23* (July 2006), <http://www.proatom.ru/modules.php?name=News&file=article&s id=594> (accessed 14 June 2009).

14 The power of the commercial breeder reactor is still under discussion (1200 or 1600 MWe).

15 F. M. Mitenkov, *Thoughts about Past Experiences* (Moscow: IzdAt, 2004). This is a very rough estimate. Official figures were never published.

16 "Engineering of NPP Unit with Fast Neutron Reactor *BN-800"*, op. cit.

17 V. Kostin, et al., "Main Ways to Improve Technical And Economic Characteristics of Nuclear Power Plants with Fast Sodium-Cooled Reactors" (Proceedings of the 2nd International Conference on the Development of the Nuclear Power Industry Based on a Closed Fuel Cycle with Fast Reactors, Rosenergoatom Conference, Moscow, 28-29 November 2007).

18 Neutron fluence is the cumulative number of neutrons that have passed through an area.

19 F. M. Mitenkov, "Perspectives of Fast Neutrons Reactors Development," *Atomic Energy 92* (2002): 423-432.

20 L. Kochetkov, op. cit.

21 L.G. Osipenko, et al. *Nuclear Underwater Epic: Heroism, Failures, Disasters* (Moscow: A/O Borges, 1994), 207-210.

22 Rosatom Nuclear Energy State Corporation (in Russian–Росатом) is a State Corporation in Russia, and operates Russia's nuclear complex.

23 V. Stepanov, et al., "Nuclear Power Unit Based on *SVBR-75/100* Reactor. The Project and Ways of its Implementation" (Proceedings of the 2nd International Conference on the Development of the Nuclear Power Industry Based on a Closed Fuel Cycle with Fast Reactors, Rosenergoatom, Moscow: November 28-29, 2007).

24 A. Bychkov, interview published electronically by ATOMINFO.RU (in Russian), <http://www.atominfo.ru/news/air4165.htm> (accessed 14 June 2009).

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6 Fast Breeder Reactors in the United Kingdom

Walt Patterson

Margaret Gowing's masterly official history of postwar nuclear activities in Britain, *Independence and Deterrence*, describes disputes among the nuclear physicists at the Atomic Energy Research Establishment at Harwell:[1]

> The only point on which there was general agreement ... was on the long term future on the ultimate and overriding importance of breeder reactors, which would produce more secondary fuel than the primary fuel they consumed.

The reason for this island of unanimity amid the prevailing conflict of views was straightforward. In the late 1940s and early 1950s uranium was scarce and expensive; moreover its supply was politically acutely sensitive, because of the weapons implications.

In consequence, as Harwell director Sir John Cockcroft explained:

> ... we have to develop a new type of atomic pile (reactor) known as the 'breeder pile' because it breeds secondary fuel (plutonium) as fast or faster than it burns the primary fuel uranium-235 ....These piles present difficult technical problems, and may take a considerable time to develop into reliable power units. Their operation also involve difficult chemical engineering operations in the separation of the secondary fuel from the primary fuel.

By 1953, nuclear engineers at Risley, after working for some two years with their Harwell colleagues on the design of a fullscale fast breeder power station, concluded:

Edited, abridged and updated from *Going Critical: An Unofficial History of British Nuclear Power*, Walter C. Patterson, Paladin, 1985.

> At first sight this fast reactor scheme appears unrealistic. On closer examination it appears fantastic. It might well be argued that it could never become a serious engineering proposition.

Nevertheless, in March 1955, construction work started on an experimental fast breeder power station at the new Atomic Energy Authority's (AEA) Dounreay Experimental Reactor Establishment, on the north coast of Scotland. This remote location was chosen precisely for its remoteness, because of major questions about the possible behaviour and misbehaviour of a reactor whose core contained an unprecedented concentration of fissile material.

By the third AEA annual report, in mid-1957, the *Dounreay Fast Reactor (DFR)* was "expected to start operation in 1958". From 8 October 1957, however, the AEA was preoccupied with the aftermath of the Windscale fire, in which one of its graphite-moderated plutonium production reactors had burned and dispersed a large amount of radioactivity. The *DFR* did not actually go critical until November 1959. The 1960 annual report remarked, "A prototype power producing reactor may be built for operation about the year 1967, the development of which will enable a commercial power station to be specified."

The design output of the *DFR* was to be 60 megawatts of heat (MWt), or 14 megawatts of electricity (MWe). Successive AEA annual reports stressed that the *DFR* was "experimental," "intended to develop the technology of fast reactors generally." It fulfilled this role admirably, in that it succumbed to a fascinating variety of novel engineering difficulties, particularly those arising from the use of molten-sodium-potassium alloy as the cooling fluid. By mid-1961 its highest output had been 1.5 MWt. By mid-December the reactor had been run up to 11 MWt, at which point it was shut down to have its fuel core replaced with one of improved design. While thus busy with the *DFR*, the AEA in 1961-62 was also completing a design study for a 500 megawatt (MW) fast breeder power station. The next step in the program would be to try out the concepts on an intermediate scale, on what would be known as the *Prototype Fast Reactor (PFR)*. The *DFR* reached an output of 30 MWt, half its intended design output, on 7 August 1962, and remained at this level for the rest of the year. In October it supplied electricity to the national grid for the first time.

The Select Committee on Nationalized Industries, in its May 1963 report on the electricity supply industry, noted that "the development by the Atomic Energy Authority of a fast breeder reactor at Dounreay ... remains a long term project. The Authority hopes that a prototype will be operating by 1969 or 1970; and the first civil station would not be working before 1975." The 1963-64 annual report of the AEA declared that "Consortia design engineers are engaged on a design study of a 1000 MWe power producing fast reactor." At the time, the largest thermal reactors contemplated for construction were 660 MWe. In July 1963 the *DFR* at last attained its full design output of 60 MWt, or 14 MWe, and operated at this level for most of the ensuing year.

In 1964-65, the AEA completed two design studies for fast breeders. The first was for the proposed *PFR*. It was to have an output of 600 MWt or 250 MWe; but it was designed to use components suitable for a full scale commercial fast breeder power station. By August 1965 AEA staff were already preparing detailed designs and specifications for major plant and civil engineering contracts for *PFR*. This was well before the official go-ahead for *PFR*, which did not come until 9 February 1966.

The AEA Annual Report 1965-66 reported that:

> The inner and outer breeder sections of the *DFR* were originally loaded in 1958 with natural uranium elements clad in stainless steel. Early in 1965 it was found that a few of the lightly irradiated elements in the outer breeder were difficult to remove, although the inner breeder elements were in good condition. A comprehensive survey of the outer breeder was carried out in September 1965, and a number of elements were found to be distorted or swollen. Investigation showed that this had been caused by higher than normal uranium temperatures due to abnormal coolant flow conditions in some regions of the breeder. This will not occur in future fast reactors since coolant flow conditions will be different, and the breeder fuel itself will be ceramic and therefore not subject to the temperature limitations of natural uranium. It was decided to remove 500 breeder elements, and to carry out the work. Special cutting tools and removal equipment had to be manufactured. The work was completed by the end of December and the reactor went critical again on 23 January after loading new experiments.

Such incidents in no way weakened AEA confidence in the concept of the fast breeder. On the contrary: while they pressed on with detailed designs for the *PFR* they had already satisfied themselves that the prospects were excellent:

> The design study of a 2x1000 MWe fast reactor power station in general endorsed the conceptual design of the *prototype fast reactor* as representing the most likely features of the first commercial fast reactors. A capital cost estimate for this study indicates a cost similar to that of the best thermal reactor available at the same time, with potential for further reductions. (AEA Annual Report 1965-66, paragraph 157)

The Select Committee on Science and Technology was duly impressed. In its first report, in October 1967, it noted that:

> The fast 'breeder' reactor is the system on which the long term prospects of nuclear energy generation are based ... Work on this system has been increasing steadily for some ten years and the greatest effort of the

AEA's research and development programme is now devoted to this type of reactor. Expenditure in 1966-67 was approximately £12 million and there will be increasing capital expenditure over the next few years as the construction of the *DFR* prototype proceeds. This system is regarded as likely to provide a very cheap source of electricity. Building costs (at 1967 prices) of fast reactor stations are expected to be as low as £50 per kilowatt installed and generating costs to be reduced ultimately to 0.3d (old pence) per kilowatt hour. The prototype, a large station producing 250 MWe, is expected to be on power in 1971.

By 1968, the AEA was looking to have at least 15 gigawatt electric (GWe) of fast breeders in operation by 1986. On the basis of "another bold decision" by government, exploitation of the fast breeder would be "the major event of the rest of the century". By 1969 the AEA was asserting that "the UK has the firm intention of introducing fast reactors as rapidly as possible after the operation of our 250 MW prototype."

Meanwhile, in May 1967, the primary cooling circuit of the *DFR* sprang a leak of molten-sodium-potassium. The reactor was shut down in July 1967 for nearly a year. The *DFR* was also manifesting other engineering problems. No reactor hitherto in operation had subjected its structural materials to intense high energy neutron radiation for lengthy periods. To find materials able to withstand the demanding environment in the core of a fast reactor was a daunting challenge.

While these practical problems occupied the attention of the staff at Dounreay, the AEA was linking up with the Central Electricity Generation Board (CEGB) and the two reactor building consortia for further design studies on commercial fast breeder power stations. On 14 October 1970, introducing the AEA annual report, chairman Sir John Hill characterized the outcome thus:

> ...we have had a most useful study of the fast reactor by a group consisting of engineers of the CEGB, the industrial design and construction firms and the Authority ... we have now an agreed programme ... which could lead to the CEGB being able to start the construction of the first civil fast reactor, possibly of 1300 MW, by early 1974 ... seeing how the *prototype fast reactor* performs in 1972 and 1973.

As it turned out, the *PFR* did not perform at all in 1972 or 1973. Nevertheless, with the *PFR* falling steadily farther behind schedule, the 1971 AEA annual report was still confident. The cooperative study had resulted in:

> the formulation of a strategic plan for the introduction of fast reactors to the CEGB network; this assumes that construction of a first commercial station will start in 1974 as a 'lead' station, following operation of the *PFR*. This would be followed by other stations after an

interval of perhaps two years. This plan assumes that the technical and economic results from the development programme confirm present expectations; it will be reviewed each year in the light of progress achieved.

In October 1971 the AEA's "present expectations" were robust:

It is estimated that, in only thirty years from now, over three quarters of all electricity in the United Kingdom will be generated from nuclear power and that more than half of this nuclear generation will stem from fast breeder reactors (to the development of which almost half the effort on the Authority's reactor programme is currently geared).

Sir John Hill had already expounded to the fourth U.N. Conference on the Peaceful Uses of Atomic Energy in Geneva in September 1971 on the "strategic plan" endorsed by the electricity authorities, the nuclear power industry and the AEA:

By 1979-80 we should have had seven years' operating experience with the *PFR*, constructional experience of perhaps three or four large commercial stations, and initial generating experience from the first of these larger units. On this basis we would expect that by approximately 1980 we would have sufficient confidence and experience to incorporate fast reactors into the United Kingdom generating system to the maximum extent consistent with the availability of fissile material and the growth of demand for new generating plant. Whether such a timetable can, in fact, be achieved will depend on technical developments over the next few years. This, however, is the plan to which we are working and so far we see no reason why it should not be achieved.

On 5 October 1972, introducing the Annual Report, Sir John Hill said:

We expect the reactor to be producing electricity by the end of 1973. We in the Authority have never proposed that the first commercial fast reactor should not be started until sufficient operating experience of the *prototype* had been obtained, to be absolutely sure that there were no fundamental problems unresolved. I have, however, always believed in continuity of design and experience and would like to see the next reactor started as soon as the lessons of the first have been fully assimilated by the designers and engineers. Clearly our hopes of a 1974 start are now too optimistic in the light of the commissioning and operating dates for the *prototype* and the amount of component testing now judged necessary. The design of the CFR is, however, under way ...

CFR was the latest acronym, standing for Commercial Fast Reactor.

The February 1973 issue of *Atom*, the AEA monthly, reported on a meeting the previous November, attended by senior civil servants and nuclear industry management, on "Future Prospects for Energy Supply and Demand," presented by the "New Systems Forum" of the AEA. According to the report of the meeting:

> A commercial fast breeder power station programme commencing with a lead station coming on line in 1981 and further stations in the mid-1980s appears to be a reasonable assumption on the basis that *PFR* knowhow and experience will be adequate for a first order to be placed for around 1976.

The almost imperceptible note of caution—1976, not 1974, and the "mid-1980s" for subsequent stations—had to be set against the assumption that a station ordered in 1976 could be "on-line in 1981." This allowed only five years for construction and commissioning, compared to the eight plus years already run up at the Dungeness B Advanced Gas-cooled Reactor station and at contemporary fossil fueled stations likewise still unfinished.

Even the faint note of caution in this report was swept aside in an aggressive presentation delivered in the United States in mid-1973 by Tom Marsham, deputy managing director of the AEA's Reactor Group:

> Satisfactory experience with the experimental reactor *DFR* in the early 1960s led to construction of the 250 MWe power station at Dounreay which will be brought to power this year. Some two or three years from then, we are expecting to start constructing the 1300 MWe lead commercial station with ordering of subsequent commercial plants building up to large scale during the early 1980s. There is nothing adventurous or foolhardy about this plan.

Nothing, perhaps, except its central premise. The end of 1973 arrived and departed with the *PFR* still awaiting its first criticality, to say nothing of being brought to power. One primary and one secondary sodium pump malfunctioned during tests. Both had to be removed from the reactor for detailed examination. Tests continued with the remaining two primary pumps. On 11-14 March 1974, however, the British Nuclear Energy Society was to play host to a major international conference on "Fast Reactor Power Stations," with delegates from France, the United States, the rest of Europe, Third World countries and even the Soviet Union. The ignominy of welcoming the foreign visitors to the conference with the *PFR* still cold was too much to contemplate. The week before the conference the AEA pulled out the control rods at Dounreay, and on 3 March 1974 started up their new reactor for the first time. On the opening day of the

conference they announced the fact with pride; it was far from coincidental that their French colleagues announced, on the closing day of the conference, that the French *Phénix* fast breeder had just attained full power.

One paper in particular, by Eric Carpenter, head of reactor physics at the CEGB's Berkeley Nuclear Laboratories, warned that the CEGB was less enthusiastic than the AEA about a rapid move into fast breeders. Reliability was crucial; together with delays in construction, lack of reliability had "a much bigger deleterious influence on electricity costs than almost any of the advantages claimed in the brochure assessments." The CEGB by this time had all too much firsthand experience of both delays and unreliability of its conventional nuclear stations, and of what the paper scornfully called "brochure assessments." The paper asserted that the putative savings from introducing fast breeders as fast as possible would be no more than 5 percent of total expenditure on a nuclear system and then only in what it called "the unlikely event of capital costs of fast and thermal reactors being equal." The CEGB contributors considered that no order for a fast breeder power station could be placed before 1977 or 1978 at the very earliest.

Throughout much of 1974, staff at Dounreay continued running the *PFR* at low power. Small leaks appeared in the steam generators, the boilers in which hot molten-sodium passed through thousands of fine tubes to boil the water around the tubes. Such leaks were a particular problem in a sodium-cooled system because of sodium's reactivity with water. A major leak, like one that had happened at the Soviet fast breeder prototype at Shevchenko in November 1973, would release enough hydrogen and heat to create a serious hazard of explosion. Even a minute leak, invisible to the naked eye, would lead to the formation of hydrogen bubbles in the sodium coolant, presenting at the very least an unwelcome irregularity in the coolant flow, and possibly actual control problems. By the end of October 1974, the most troublesome steam generator was decoupled from the reactor in order to find the leaking tube and plug it.

Six months later, the *PFR* once again played host to a visiting party. At the end of April the newly formed European Nuclear Society (ENS) held its inaugural conference in Paris. After the conference, one of the side trips took participants from all over Europe to Dounreay. AEA staff were happy to show off their reactor, which was, they said, working fine; a month earlier the plant had generated its first electricity. Unfortunately, however, it had yet to reach a power level above 12 percent of its full thermal capacity. Small but persistent leaks in the sodium water steam generators kept two of the reactor's three cooling circuits out of operation. *PFR* staff carried on operating the reactor on its one remaining cooling circuit, but trouble with turbine bearings interrupted even this limited operation. Then, just before the nuclear dignitaries arrived from Paris, more small leaks manifested themselves, this time in a section of the only operative cooling circuit.

The AEA staff at Dounreay put on brave faces, but the ENS visit cannot have been an especially happy occasion. As the editor of *Nuclear Engineering International*, put it: "Although the reactor itself has been operating very well it has not yet been possible to build up any significant amount of fuel irradiation." Nor, it might be added, to generate any significant amount of electricity. The AEA continued to protest that the reactor itself was working well, and that the stubborn troubles at Dounreay were with the generating set and the steam generators. But the CEGB had already suffered many years of frustration with its own generating sets, and knew what a headache these could be.

Furthermore, to suggest, as the AEA was trying to, that the steam generators were somehow ancillary, not part of the nuclear system, was indefensible special pleading. One of the unique distinguishing characteristics of the fast breeder design selected by the AEA was precisely the choice of molten-sodium as a coolant. If you could not then use the molten-sodium reliably to boil water, you had a basic design problem – one that could not be brushed aside by reference to the satisfactory operation of the reactor core itself.

In February 1976, *Nuclear Engineering International* was blunt:

> Hope that the Dounreay *Prototype Fast Reactor* (*PFR*) would be brought up to full power in February will not now be fulfilled. The designed output of 250 MWe is not now likely to be achieved 'for several months'. The reactor continues to operate satisfactorily and with number 1 secondary (cooling) circuit in operation an electrical output of 40 MWe has been achieved with a thermal power of approximately 200 MW (of heat) ... Work in preparation for recommissioning of number 3 secondary circuit is well advanced. The circuit has been filled with sodium and cleanup operations are in progress ... It was expected that this circuit would be available for power operation during the next few weeks. On number 2 circuit, work on checking the superheater and to determine how best to operate has progressed well.

By September 1976 some of the news from Dounreay, as noted in *Nuclear Engineering International*, was at last genuinely good:

> During most of August the 250 MWe *PFR* at Dounreay has been operating on all three of its coolant loops with all of the early heat exchanger problems now remedied. The maximum power reached so far is 500 MWt, but full power was expected to be reached by the first week in September.

The report continued, however, with additional news of a slightly more disconcerting kind:

Plans to replace all three types of heat exchanger with improved designs using austenitic steel and avoiding the thick tube plates where corrosion has occurred are still proceeding as scheduled for installation in 1979.

When this schedule for replacing major plant components with completely new ones had been decided, the magazine did not say. It was nevertheless a further indication that the *PFR* was a long way from demonstrating that fast breeders could fulfil the CEGB's requirements that they be reliable, built on schedule and within budget. The AEA said that the replacement heat exchangers would be in service by 1979. They were not. Over the years, periodic questions in Parliament elicited monotonously similar answers: the cumulative capacity factor (output of electricity from the *PFR* as a fraction of its design capacity) remained stuck year after year at approximately 10 percent. In October 1984, the authoritative quarterly analysis published in *Nuclear Engineering International* gave the total lifetime capacity factor of the *PFR* in the first ten years after its startup as 9.9 percent.

On 23 March 1977 Lord Hinton, who had chosen the Dounreay site and supervised the early stages of construction of the *DFR*, threw the switch that consigned it to history. His reflective remarks on the occasion, reprinted in the AEA monthly *Atom*, were a tour de force of personal reminiscence interspersed with incisive views on the current state of the art, including the *PFR*:

> I hope and believe that many lessons have been learned from *PFR*. At one of the early Fast Reactor Design Committee meetings Jim Kendal, whose feet were usually very firmly on the ground, put forward a complicated proposal for the design of the fast reactor and I remember saying to him, 'Look Jim, that's a very clever idea but I don't pay you to be clever, I pay you to be successful'. Most of the mistakes (and fortunately they have been rectifiable) on *PFR* have been made because engineers have thought they were just that little bit more clever than any of us really are.

Hinton went on to endorse the proposal to build a full scale fast breeder "not later than the end of this year ... the aim should be to commission it before 1985." Unfortunately, however, Hinton's assumption about the ready rectification of the mistakes on the *PFR* was premature.

Another Dounreay mistake was to dump an assortment of discarded material, much of it uncatalogued and unrecorded, into a disused access shaft leading into a waste-disposal tunnel under the seabed offshore. On 10 May 1977 an explosion in the shaft blew its five-tonne concrete cap off and scattered debris in all directions. Investigations suggested that waste contaminated with sodium-potassium coolant had produced hydrogen in the shaft. The explosion happened less than a month before the opening of the intensely controversial public inquiry into the proposed

Thermal Oxide Reprocessing Plant (THORP) at what was then called Windscale. A key reason for THORP was to recover plutonium for the U.K.'s long-anticipated fast breeder power stations. Perhaps not surprisingly, almost no word about the Dounreay shaft explosion reached the media at the time.

The U.K. commitment to reprocessing was based on the assumed rapid commercialization of fast reactors. From the mid-1960s official U.K. opinion, led by the AEA, assumed that a rapid progression from the little *DFR* to the larger *PFR* to a series of full scale fast breeder power stations was not only natural but obviously desirable. The only possible constraint foreseen was a conceivable shortage of plutonium to fuel the full scale fast breeders. With that in mind, the reiterated policy of Government and AEA was to reserve all "civil" plutonium separated from U.K. spent fuel, against its imminent use to fuel the coming fast breeder power stations. Even in 1975, when the *PFR* had at long last gone critical only to manifest the sodium leaks that would cripple it, the official commitment remained unshaken.

A measure of this commitment could be seen from the AEA's evidence to the Royal Commission on Environmental Pollution, chaired by Sir Brian Flowers. In September 1975, the AEA submitted a paper to the Flowers Commission taking as its premise a nuclear programme that would have a total of 104 GWe of nuclear power in operation by the year 2000, of which no less than 33 GWe would be fast breeders. At the time the total operative nuclear generating capacity in Britain was less than 5 GWe, the nuclear plant construction industry was in chaos and the *PFR* had yet to attain more than a modest fraction of its intended design output. Sir Brian Flowers, himself a part time board member of the AEA, was reported to have taken exception to this scenario as being utterly unreal. The AEA insisted that it was not a forecast, merely a "reference programme" to establish an upper limit on the scale of British nuclear involvement for purposes of weighing environmental impact. Be that as it might, the AEA clearly considered this "reference programme" as achievable.

Since the beginning of the 1970s, the AEA had been pleading for government permission to build its long-awaited Commercial Fast Reactor. Design teams from the AEA, the CEGB and the nuclear plant manufacturers had been busying themselves for years laying out their paper power plant, based on a 1.2 GWe fast breeder. By 1976, the AEA was spending close to £100 million a year in funding on research and development on the fast breeder. In 1976, confident rumour had it that the go-ahead for the *CFR* was at last imminent.

The rumour had received a boost from the suggestion that the Flowers Commission would be advocating the CFR. At the end of 1975, however, Sir Brian Flowers declared that this suggestion was "quite false." Flowers published letters he had exchanged with Prime Minister James Callaghan, asking that the Government hold off any decision "on whether to proceed with such a plant in collaboration with other European countries" until after the Commission published its report

some months later. Failing such a postponement, the Commission wanted to see a clear distinction drawn between a single full scale demonstration fast breeder and a large continuing programme of such plants. The Commission conceded that, by building one full scale plant Britain might contribute significantly to resolving what the Commission called "the serious fundamental difficulties" associated with the fast breeder. No official body had for many years so much as hinted that the fast breeder could even raise "serious fundamental difficulties." Flowers indicated indirectly in his letter what these difficulties might be:

> The demonstration site should be remotely sited; it should have its own fuel reprocessing and fabrication plant on site in order to remove the security risks of shipment of plutonium; it should be provided with every means of protection, including both physical devices and an armed security force; and experience of plutonium accountability and inspection should be designed into its system.

It was not exactly a reassuring recipe.

On 22 June 1976, at Energy Secretary Tony Benn's National Energy Conference, Flowers was more specific about the Commission's unease about the use of plutonium as a civil fuel. Earlier in June, Benn had told the Commons that the Government would announce in the early autumn its decision about the future of Britain's fast breeder programme. Work had reached a point at which the Government had to decide:

> our approach to the next stage of the system's development, including our policy on the construction of a fullscale demonstration reactor. This is a matter of great public importance in terms of long term energy provision and the safety and environmental considerations. In my current review of this I wish to provide the opportunity for wide consultation. I shall take full account of the prospects for international cooperation and the forthcoming report on radiological safety from the Royal Commission on Environmental Pollution.

As it turned out, however, the Royal Commission was concerned about more than just radiological safety. The Commission was deeply apprehensive about the implications of a commitment to what it called the "plutonium economy." This term was coined by Glen Seaborg, a co-discoverer of plutonium, chairman of the U.S. Atomic Energy Commission during most of the 1960s, and an enthusiastic booster of plutonium-breeder reactors. The Royal Commission accepted that there was a case to be made for building a single large fast breeder to assess its safety and social implications. But the Commission went on to warn that "we must view this highly significant first step with misgivings ... The strategy that we should prefer to see adopted, purely on environmental grounds, is to delay the

development of CFR1" (paragraphs 517-18). After the publication of the Flowers report, on 22 September 1976, the prospect for even a single large fast breeder in Britain became distinctly bleaker.

In September 1977 the Select Committee on Science and Technology published the report of its study into so called "alternative sources of energy." AEA chairman Sir John Hill welcomed the committee's recommendation that CFR be built. The following month, at a Royal Institution conference cosponsored by nuclear proponents and opponents, Sir Brian Flowers, speaking in the role of a critic in the session on fast breeders, concurred with his co-speaker, the AEA's Tom Marsham, that one large fast breeder was indeed to be recommended. Nevertheless, despite this apparent closing of ranks within the U.K. nuclear establishment, the Government was less and less eager to give CFR the green light.

Added to this was the view expressed by Sir John Hill, that the AEA did not regard the proposed large fast breeder as in any way an experimental plant. On the contrary, it would just be another nuclear power station, of a new design. Behind this confident assertion lay a crucial corollary: if the new plant was just another power station, it would obviously be paid for not by the AEA but by the electricity suppliers, just as they paid for all their other power stations. However understandably appealing this idea was to the AEA, it did nevertheless come up against a basic problem. The CEGB did not want a fast breeder power station – not, at any rate, if it had to pay for it.

Furthermore, the AEA had by this time undermined its own position, by relabeling its proposed plant. It would be not a Commercial Fast Reactor but a CDFR (Commercial Demonstration Fast Reactor). The internal contradiction in this new label did not go unremarked: surely a plant was either commercial or a demonstration plant? The new designation amounted to an admission by the AEA that the plant would not be in any conventional sense "commercial." It would "demonstrate" the design for a commercial plant; but its electricity output would not be competitive in cost with that from conventional generating plants.

The CEGB let it be known that it would make a site available for a large fast breeder linked to the CEGB system; but it had no intention of putting up the capital for such a plant. The collapse of electricity demand growth was already embarrassing. The CEGB's excess generating capacity was headache enough as it was, without adding more: especially with the probable aggravation of a novel design. The AEA might get away with pronouncing itself pleased because the *PFR*'s reactor itself was working properly, despite the deep seated troubles with the steam generators. The CEGB could not take such consolation.

The OPEC oil shock in 1974 had triggered an economic recession throughout the industrialized world. Soaring fuel prices stunned energy users into a new and thriftier awareness of their previous extravagance. Electricity consumption

stopped increasing. In some countries like Britain it even decreased. Interest rates in double figures made nuclear power, with its huge capital costs, even less competitive with conventional fuels. The grandiose global vision of an energy future centered on plutonium fueled fast breeders began to look less and less plausible.

From 1978 onwards, official support for introducing the pressurized water reactor (PWR) to succeed the United Kingdom's gas-cooled graphite-moderated reactors was also tacitly sidelining the fast breeder. Nevertheless the election of the Conservatives under Margaret Thatcher in 1979 noticeably revitalized official support for fast breeders; one of Mrs. Thatcher's first official visits was to Dounreay, on 6 September 1979. In 1981-82 the focus of nuclear controversy was the battle over the pressurized water reactor at Sizewell B. The fast breeder people kept their heads down.

On 29 November 1982, the Secretary of State for Energy, Nigel Lawson, told the House of Commons that:

> The Government has now completed its review of the Fast Reactor (initial capitals in the original). The Fast Reactor is of major strategic significance for the U.K.'s and the world's future energy supplies. It ... can create out of the spent fuel and depleted uranium which has so far arisen from our thermal programme fuel equivalent to our economically recoverable coal reserves.

> The UK is among the world's leaders in the development of this technology. Through the successful programme of research and development undertaken by the Atomic Energy Authority, which centers on the operation of the *Prototype Fast Reactor* and associated fuel cycle at Dounreay, we have demonstrated the feasibility and potential of this technology ... The Government has therefore decided to continue with a substantial development programme for the fast reactor based on Dounreay ...

However, Lawson then continued:

> ...we now believe that the series ordering phase will begin in the earlier part of the next century ... the development programme will be geared to this timescale ... The Government and the Atomic Energy Authority have been having exploratory discussions with other countries to establish ... the potential for collaborating with other countries as a means of securing the maximum benefits from this vital development programme.

Not everyone was convinced. Even *Nuclear Engineering International* had doubts. In February 1983, it declared:

> The large amounts of money being spent worldwide by the nuclear industry on the development of fast breeder reactors is becoming increasingly difficult to justify... Will it ever be possible to recoup the vast sums that have been spent and the much greater sums that will need to be spent before the fast reactor can become a commercial option for electricity utilities? ... Uranium will not be suddenly exhausted or become excessively expensive ... There will be plenty of time to identify the trend ... But perhaps of greater significance to fast reactor economics than the availability of uranium is the fact that with advances in techniques for the storage of irradiated fuel from light-water reactors utilities can avoid reprocessing. The uncertain and growing costs of reprocessing are then properly loaded on the fast reactor ... In these circumstances fast reactors may never be economic ... Evangelical fervour is not a substitute for sound technical argument.

In February 1984 the Comptroller and Auditor General published a terse report entitled "Development of Nuclear Power," expressing unease about the AEA's financial performance; and the House of Commons Committee of Public Accounts looked into the matter. The committee chairman asked AEA chairman Sir Peter Hirsch "the estimated total cost of development" of the fast breeder. Sir Peter replied: "We have spent so far about £2400 million in 1982-83 prices. The forward development programme, assuming a certain profit for it, again in 1982-83 prices, is estimated to be £1300 million, the total being £3700 million." Asked "What have you got for all this money?" Hirsch continued:

> The main thing we have got is that we have got the expertise in the UK to go forward to build a CDFR and then have a commercial programme. For that money we shall be, we are, in the position to give the UK the option of having a fast reactor capability for producing electricity. We have done a cost benefit analysis of what the country would get out of it, making certain assumptions. Assuming that commercialization of the fast reactor starts in about 2015 and you have a programme of building fast reactors of 1.25 gigawatts electrical for about 30 years, you can estimate, admittedly on making certain assumptions of uranium price escalation, that you would expect benefits of several billions of pounds compared to the cost you would have to pay if you got the electricity from PWRs ...

On 19 July 1984, the Select Committee on Energy pointed out the real import of Hirsch's evidence:

Since 1955-56 some £2400m (in 1982-83 money values) has been voted for fast reactor R&D, and in the twenty years since 1962-63 real expenditure has remained remarkably steady at between £85m and £120m a year. In evidence to the Committee of Public Accounts on 2 April 1984, the Chairman of the UKAEA estimated that a further 25-30 years and additional R&D expenditure of £1300m (in 1982-83 prices) will be needed to reach the stage 'where one hopes to obtain a commercial station'. To this figure must be added £2 billion construction costs for a commercial demonstration reactor and £300 million for reprocessing facilities, giving total estimated further expenditure of £3.3 billion and a cumulative figure of £5.7 billion. This implies that at present the fast reactor is roughly halfway through a perceived 60-year research, development and demonstration programme...

It did not get much farther.

Public concern about the health effects of Dounreay were growing. In 1983 radioactive particles of spent fuel were found on adjacent beaches. How they got there has never been established; but investigations discovered a plume of radioactivity in the sea fanning out from the site. Late in 1985, even as the AEA was promoting a plan for a European Demonstration Reprocessing Plant for fast breeder fuel at Dounreay, the Thatcher Government cut funding for fast reactor development. Then, on 26 April 1986, came Chernobyl. The accident cast a pall over every form of nuclear activity. Public opposition erupted, even at Dounreay. Then yet another steam-generator failure shut down *PFR* for six months.

On 21 July 1988, minister Cecil Parkinson announced in the House of Commons that annual funding for fast breeders was to be cut from £105 million to £10 million, that funding for the *PFR* would cease after 1994 and for Dounreay reprocessing after 1997. It was the death knell for the U.K. fast breeder programme. After four decades of effort, and public expenditure of over £2400 million, it had proved to be a radioactive dead end.

Two decades after Parkinson's announcement, the cleanup of Dounreay continues, as does the drain on public funds. The once all-powerful AEA, broken up and sold to private interests, is a shadow of its former self. But work at Dounreay will last for decades to come. Decommissioning the *PFR*, dealing with the now-notorious shaft, clearing up ponds and other facilities and decontaminating the site will last into the 2030s and beyond, at a cost as yet difficult to determine. Looming in the background is one further question. The collapse of the fast breeder program leaves the U.K. with an inventory of separated plutonium amounting to about 100 tonnes. What is to become of it? No one in Government is saying – probably because no one knows.

Endnotes

1. Margaret Gowing and Lorna Arnold, *Independence and Deterrence: Britain and Atomic Energy, l945-52*, 2 Vols. (London: Macmillan, l974).

7 Fast Reactor Development in the United States

Thomas B. Cochran, Harold A. Feiveson, and Frank von Hippel

Immediately after the bombing of Pearl Harbor on December 2, 1942, research on plutonium production for atomic weapons was consolidated at the University of Chicago under Nobel Laureate Arthur H. Compton. The "Metallurgical Laboratory" (later to become Argonne National Laboratory) was the code name given to Compton's facility. It was here that a small group of scientists, led by Enrico Fermi, built the world's first reactor, *Chicago Pile-1* (*CP-1*), which achieved initial criticality on 2 December 1942. During the next two years, work on the development of plutonium production reactors shifted to Oak Ridge and then Hanford. By early 1944, Compton and the Chicago scientists began thinking about the role of the Metallurgical Laboratory after the war.[1]

On the morning of April 26, 1944, Enrico Fermi, Leo Szilard, Eugene Wigner, Alvin Weinberg and others gathered to discuss the possibilities for using nuclear fission to heat and light cities.[2] The scarcity of fissile material was on everyone's mind. It was unclear at that time whether there was sufficient uranium even for producing highly enriched uranium and plutonium for a significant number of nuclear weapons. Fermi and his colleagues at the Metallurgical Laboratory therefore cast around for ways to produce maximum power — or plutonium for weapons — with minimal resources.[3] They recognized that some reactor configurations might permit the conversion of uranium-238 to fissile (chain-reacting) plutonium at a rate faster than the fissile uranium-235 was consumed, hence the term "breeder reactor."

Walter Zinn, one of the nation's few reactor experts and a close colleague of Fermi, was soon recruited to the cause.[4] By summer of 1944 he had begun a more detailed investigation of breeder reactor designs. By the end of 1945, he had abandoned the idea of breeding uranium-233 in thorium and confirmed the original plan of breeding plutonium-239 from uranium-238 using fast fission neutrons.[5] In 1945 Enrico Fermi said, "The country which first develops a breeder reactor will have a great competitive advantage in atomic energy."[6]

The world's first fast-neutron reactor was *Clementine*, a 25 kilowatt thermal (KWt), mercury-cooled experimental reactor built at Omega Site (TA-2) at Los Alamos.[7] It was proposed and approved in 1945. High intensities of fission-spectrum neutrons were needed by the bomb designers. Also, operation of the reactor would supply information about fast reactors that would be relevant to their possible use for production of power and fissile materials.[8]

Construction began in August 1946, criticality was achieved in late-1946, and full power in March 1949. The fuel was plutonium metal with natural uranium slugs at each end of the steel-clad rods. The rods were installed in a steel cage through which the liquid-mercury coolant flowed, driven by an electromagnetic pump. The core was surrounded concentrically with a 15 cm thick natural uranium reflector, a 15 cm thick steel reflector and a 10 cm thick lead shield.[9]

Clementine was shut down in March 1950 due to a control rod malfunction. Operations resumed in September 1952. It operated only until 24 December 1952, however, when a fuel rod ruptured. The uranium slugs swelled, burst the cladding and released plutonium into the mercury coolant.[10] The reactor was subsequently dismantled.

After *Clementine*, Los Alamos developed and briefly operated one additional fast reactor, *LAMPRE-I*. This sodium-cooled reactor was fueled with molten plutonium. It achieved initial criticality in early-1961 and operated successfully for several thousand hours until mid-1963. Designed to explore issues associated with using plutonium fuel in fast breeder reactors, it was originally intended to operate at 20 megawatt thermal (MWt). It became apparent, however, that knowledge was inadequate about the behavior of some of the core materials in a high-temperature, high-radiation environment.[11] The design power therefore was reduced to 1 MWt, with the plan to follow *LAMPRE-I* by a 20 MWt *LAMPRE-II*. By mid-1963, *LAMPRE-I* had served its intended purpose and was shut down. Funding for the construction of *LAMPRE-II* never materialized.[12]

Admiral Hyman G. Rickover briefly experimented with fast-neutron reactors for naval submarine propulsion. This effort began with General Electric's development and operation for the Navy of the land-based *S1G* prototype at the Knolls Atomic Power Laboratory in West Milton, New York. The *S1G*, which was HEU-fueled, operated from the spring of 1955 until it was shut down in 1957 after Admiral Rickover abandoned fast reactors for naval propulsion. During its brief operating history, the sodium-cooled *S1G* experienced trouble with leaks in its steam generators.[13]

The *S1G* prototype was followed by the deployment of the *S2G* fast reactor in the nuclear submarine, USS Seawolf (SSN 575). According to Atomic Energy Commission (AEC) historians, Hewlett and Duncan, in their history of the U.S. nuclear navy from 1946 to 1962:

Although makeshift repairs permitted the Seawolf to complete her initial sea trials on reduced power in February 1957, Rickover had already decided to abandon the sodium-cooled reactor. Early in November 1956, he informed the Commission that he would take steps toward replacing the reactor in the Seawolf with a water-cooled plant similar to that in the Nautilus. The leaks in the Seawolf steam plant were an important factor in the decision but even more persuasive were the inherent limitations in sodium-cooled systems. In Rickover's words they were 'expensive to build, complex to operate, susceptible to prolong shutdown as a result of even minor malfunctions, and difficult and time-consuming to repair.'[14]

Consolidation of breeder reactor research at Argonne National Laboratory

In 1946, the newly formed AEC took control of the nation's nuclear research facilities and tapped Zinn to head the Chicago laboratory, which by then had been reorganized and renamed Argonne National Laboratory (ANL). The next year, the AEC Commissioners decided to consolidate the entire AEC reactor program at ANL.[15] The Commission needed reactors not only to produce plutonium for weapons but also for the production of radioisotopes and for general research. There was also widespread public interest in using reactors to generate electric power.[16]

In drafting his section of the General Advisory Committee report, Zinn stressed power reactors. Here (as had been the case since 1944) a fact of supreme importance was the shortage of fissionable material. Existing stocks of uranium ore seemed scarcely large enough to sustain production of a modest number of weapons, to say nothing of providing fuel for power plants. Zinn believed the only hope for power reactors lay in those which would breed more fissile material than they consumed.[17]

Zinn convinced the AEC to give the breeder project a high priority and insisted on directing the effort himself. Fermi promoted it by giving lectures extolling the goal of extracting almost 100 percent of the fission energy from natural uranium.[18]

Experimental Breeder Reactor-I

On November 19, 1947, the AEC authorized ANL to design and build a liquid-metal-cooled, fast-neutron reactor, the second fast reactor in the United States, *Experimental Breeder Reactor-I (EBR-I)*, alternately known as "Chicago Pile 4" and "Zinn's Infernal Pile."

The *EBR-I* team decided to cool the reactor core with a sodium-potassium (NaK) alloy. Since they knew little about the effect of this liquid-metal coolant on materials and worried that the control rods might stick or corrode, they decided to cool them with air, which introduced the complexity of designing

two completely separate cooling systems. This was especially hard because the sodium-potassium metal would burn in both water and air. Therefore, there could be no fluid leakage.[19]

From the beginning of the Manhattan Project, questions had been raised about the public safety concerns associated with building reactors in the Chicago area. By summer 1948, Zinn was convinced the project needed to be built at a remote site and asked the AEC to find one.[20] The Commissioners chose a site near Arco, Idaho, that had been a proving ground for navy ordnance. It came to be known as the National Reactor Testing Station, now part of the Idaho National Laboratory (INL) and soon housed other ANL reactor projects as well as other government reactors.[21]

EBR-I was the first fast-neutron reactor designed to both breed plutonium and to produce electric power. The 1.2 MWt (0.2 megawatt electric)[22] sodium-cooled reactor went critical on December 20, 1951, and lit four 200-watt light bulbs, thereby becoming the world's first electricity-generating nuclear power plant. See figure 7.1. *EBR-I* was fueled with weapon-grade (94 percent-enriched) uranium. On June 4, 1953, the AEC announced that *EBR-I* had become the world's first reactor to demonstrate the breeding of plutonium from uranium.

Unfortunately the reactor was designed with a prompt positive power coefficient of reactivity (increases in power had a positive feedback). On November 29, 1955, during an experiment to obtain information about this instability, the reactor had a partial (40–50 percent) core meltdown. The damaged core was removed and the reactor was repaired and operated until it shut down on December 30, 1963.

The accident at *EBR-I* focused attention on safety issues associated with liquid-sodium fast-neutron reactors and especially the possibility of an explosive criticality due to the partial melting and collapse of the core. This possibility was first studied by Bethe and Tait.[23] By 1983, the effective end to the U.S. fast reactor commercialization program, U.S. analysts had concluded that the Bethe-Tait analysis was overly conservative regarding the magnitude of the potential energy release in a fast-reactor accident, but that there were no "universally accepted estimates of upper limits on consequences of hypothetical fast-reactor accidents."[24]

The one kilowatt (KW) ANL *Fast Source Reactor* was also built at the National Reactor Testing Station to produce neutrons for the fast reactor development program. Reactor startup occurred on October 29, 1959 and the reactor was operational until sometime in the late-1970s, when it was moved to a new location on the Idaho site.
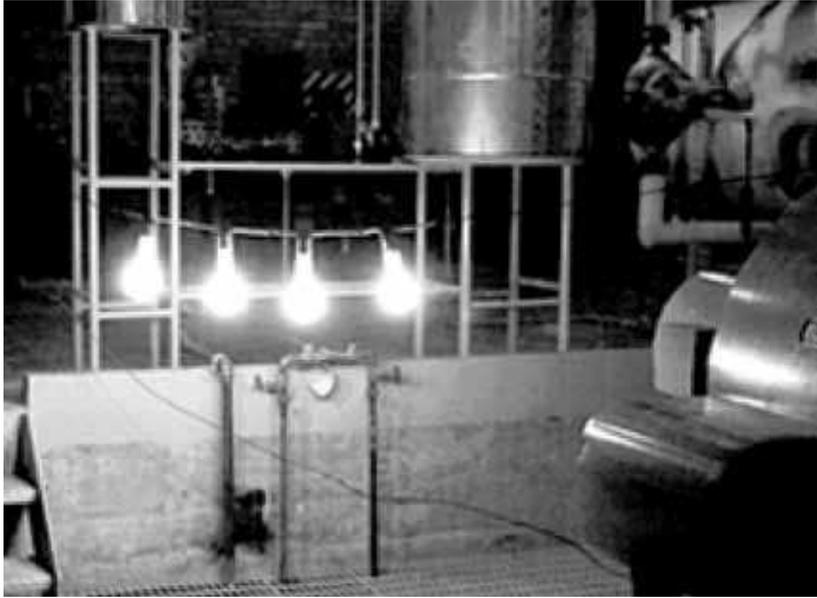
Figure 7.1 **Experimental Breeder Reactor−I.** By illuminating four light bulbs EBR−I became the world's first electricity-generating nuclear power plant on Dec. 20, 1951. Source: Argonne National Laboratory.

Experimental Breeder Reactor-II

Experimental Breeder Reactor-II (EBR-II) was arguably the most successful of the U.S. fast reactors. See figure 7.2. It was a 62.5 MWt, 20 megawatt electric (MWe), sodium-cooled, "pool-type" reactor, i.e. the heat exchangers for transferring heat to a secondary loop of liquid sodium were submerged in the reactor vessel. It was designed by ANL and constructed, beginning in June 1958, at the National Reactor Testing Station (today the Materials and Fuels Complex in the Idaho National Laboratory). Criticality at low power without sodium coolant was achieved on September 30, 1961; criticality with sodium coolant on November 11, 1963; and design power on September 25, 1969.

EBR-II demonstrated the feasibility of a sodium-cooled fast breeder reactor operating as a power plant. It operated initially with metallic HEU fuel. A hallmark feature was that it had an adjoining Fuel Cycle Facility (FCF) (now called the Fuel Conditioning Facility) that permitted continuous reprocessing and recycling of fuel to keep the working inventory down.[25] *EBR-II* spent fuel was processed and fresh fuel fabricated at the FCF from 1964 to 1969.[26] In 1967, the *EBR-II* was reoriented from a demonstration plant to an irradiation facility.

Figure 7.2 **Experimental Breeder Reactor–II.**
Source: Argonne National Laboratory.

After cancellation of the Clinch River Breeder Reactor (CRBR*)* in 1983, the *EBR-II* reactor and the FCF became the research and demonstration facilities for the Integral Fast Reactor (IFR) concept promoted by ANL. The IFR program was terminated and *EBR-II* began shutdown operations in September 1994, after 30 years of operation.

The *EBR-II* shutdown activities included defueling and draining the primary and secondary sodium loops. The FCF has been converted to a Fuel Conditioning Facility whose mission is to electrochemically treat spent *EBR-II* fuel to create radioactive waste forms that are acceptable for disposal in a national geological repository. The fuel is not considered suitable for direct disposal in a geological repository because it contains sodium to provide a good thermal link between the fuel pellets and the fuel cladding. Sodium would react with any water that penetrated the cladding to generate hydrogen. The laboratory has signed an agreement with the state of Idaho that the fuel conditioning work will be completed by 2035.

The short life of the first commercial breeder reactor – Fermi 1
The *Enrico Fermi Atomic Power Plant (Fermi 1)* was the brainchild of Walker L. Cisler, who in 1951 became president and general manager, and later CEO and chair of the board of Detroit Edison. Nuclear energy had caught Cisler's attention in 1947 when he joined an AEC advisory committee on how to make connections with private industry. In December 1951, Cisler presented to the AEC a Dow-

Detroit Edison study, one of four industry studies that found that "atomic energy had an important potential for power production even if reactors were not yet economical for that purpose alone."[27]

In 1952, Cisler assumed the leadership responsibilities for organizing electric utilities to develop the Enrico Fermi Breeder Reactor Project. The project was formally organized in 1955 as the Power Reactor Development Company (PRDC) with 34 companies participating. In January 1956, PRDC applied to the AEC for a construction permit to build the reactor on the shore of Lake Erie at Lagoona Beach (near Newport, 30 miles from Detroit), Michigan. The construction permit was granted on August 4, 1956, groundbreaking took place four days later, and the pouring of concrete began in December 1956.[28]

In terms of core size and power, the Fermi 1 reactor was the largest fast-neutron reactor built up to the time. Criticality was achieved on August 23, 1963. The 200 MWt (66 MWe) sodium-cooled HEU-fueled power reactor differed from *EBR-II* in that it was based on a loop design in which the liquid sodium primary coolant transfers its heat to secondary sodium in an external intermediate heat exchanger.[29]

In October 1966, a blockage of the flow of sodium through part of the core caused a partial core meltdown. The accident was attributed to a zirconium plate that had become unfastened and obstructed the sodium flow into a fuel assembly. Two of the 105 fuel assemblies melted during the incident, but no contamination was recorded outside the containment vessel. This accident inspired the book, *We Almost Lost Detroit*.[30]

Damage to the reactor and fuel assemblies took approximately four years to repair. In May 1970, the reactor was ready to resume operation, but a sodium explosion delayed startup until July. In October, the reactor finally reached a power level of 200 MWt. During 1971, it only generated 19.4 gigawatt-hours (GWh) of electricity, however, corresponding to an average capacity factor of 3.4 percent. The PRDC therefore declined to purchase additional uranium fuel to continue plant operation. In August of 1972, upon denial of the extension of its operating license, shutdown of the plant was initiated. Operation ended on September 22, 1972. The decision to decommission the plant was made November 27, 1972. It was officially decommissioned on December 31, 1975.

Liquid Metal Fast Breeder Development in the 1960s and 1970s

Despite the commercial failure of *Fermi 1*, the U.S. Liquid Metal Fast Breeder Reactor (LMFBR) development effort picked up momentum in the 1960s, aiming for commercialization of the breeder before the end of the century.[31] In its 1962 Report to the President on Civilian Nuclear Power, the AEC specifically recommended that future government programs include vigorous development and timely introduction of the breeder reactors, which the Commission believed

essential to long-term use of nuclear energy on a large scale.[32] By 1967, the LMFBR was the AEC's largest civilian power development program.[33] The Commission's program began to embrace efforts to build an industrial base and obtain acceptance of the LMFBR by utilities, primarily through planned government-subsidized construction of commercial-scale LMFBR power plants.[34] The Commission came to see its program "as the key to effecting the transition of the fast breeder program from the technology development stage to the point of large-scale commercial utilization."[35]

In furtherance of these objectives, the Commission, in 1968, issued a 10-volume LMFBR Program Plan prepared by ANL. The dual objectives of the plan were to:

1. Achieve, through research and development, the necessary technology; and,

2. "(A)ssure maximum development and use of a competitive, self-sustaining industrial LMFBR capability."[36]

The aim was to develop an economically viable, commercial-scale LMFBR by the mid-1980s.[37] In a 1969 cost-benefit study of the breeder program prepared by the AEC, the LMFBR commercial introduction date was assumed to be 1984.[38]

With growing concern about a possible energy crisis, rapid commercial implementation of LMFBR technology had become a national mission.[39] It would remain AEC's highest priority development program until 1977, when President Jimmy Carter sought to cancel the Demonstration CRBR project; and it remained a high priority program until 1983 when the CRBR project was terminated by Congress.

In the style of President Kennedy's 1960 commitment to put an American on the moon by the end of the decade, President Nixon, in his June 4, 1971 Energy Message to Congress, announced as the highest priority item of his energy program "(a) commitment to complete the successful demonstration of the LMFBR by 1980."[40] This goal was endorsed by Congress' Joint Committee on Atomic Energy.[41]

LMFBR program expenditures

In 1975 the U.S. Government Accountability Office (GAO) estimated that the "AEC's total LMFBR program funding through fiscal years 1948–74 was approximately $1.8 billion."[42] GAO gave the LMFBR Program costs for fiscal year 1975 as $481 million,[43] which, in 2006 dollars would be approximately $1.6 billion (figure 7.3). The commercialization effort featured two components, a base program R&D effort focused on two test reactors, and a demonstration plant effort, the CRBR.
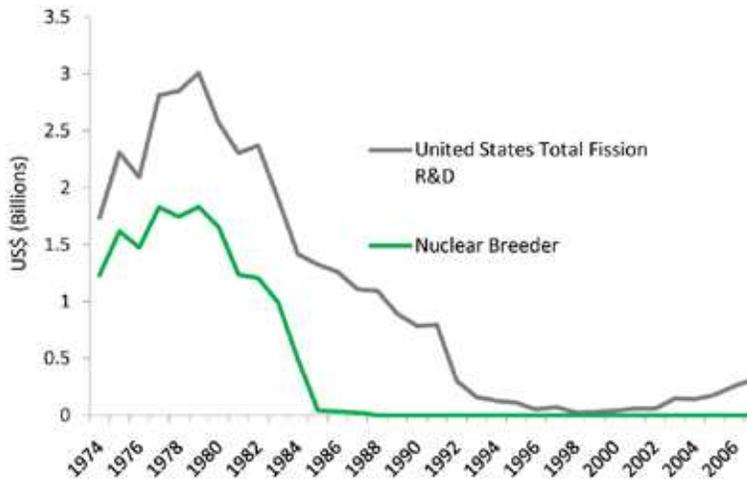
Figure 7.3 **U.S. fission R&D expenditures, 1974–2006.**
Source: International Atomic Energy Agency.

Southwest Experimental Fast Oxide Reactor

All early fast breeder concepts were based on metallic fuel. In the 1960s, however, work was begun on the use of ceramic, mixed plutonium-oxide/uranium-oxide (MOX) fuel. The *Southwest Experimental Fast Oxide Reactor (SEFOR)* was a 20 MWt sodium-cooled MOX-fueled fast-neutron reactor designed to determine the operating characteristics of a reactor with MOX fuel, and, in particular, to examine the implications of the Doppler thermal feedback coefficient associated with the use of MOX.[44] *SEFOR* did not produce electricity.

Located near Strickler, Arkansas, *SEFOR* was built and operated for the AEC by General Electric Company under the Southeast Atomic Energy Associates, a nonprofit consortium formed by 17 power companies and European nuclear agencies including the Gesellschaft für Kernforschung of Karlsruhe, West Germany.

Experiments at *SEFOR* confirmed that the negative temperature coefficient of reactivity associated with the use of mixed-oxide fuels would improve the safety of fast reactors under accident conditions involving increases in the fuel temperature.

SEFOR began operating in May 1969, and was shut down three years later. The fuel and irradiated sodium coolant were removed and taken offsite later in 1972, and some dismantling performed. The reactor was acquired by the University of Arkansas in 1975 and is still owned by the university, although the university has never operated it.[45]

Fast Flux Test Facility

It was thought by the AEC that scaling up components from existing fast reactors (*EBR-II* was 62.5 MWt and *Fermi 1* was 200 MWt) to the size of the proposed CRBR demonstration plant (975 MWt), was too risky technologically to take in one step. Therefore, an intermediate-size reactor, with a mission to test fuels, was inserted into the U.S. LMFBR development program. In July 1967, the U.S. Congress authorized construction of the *Fast Flux Test Facility (FFTF),* which at that time was estimated to cost $87.5 million and scheduled to begin full-power operation in early 1974.[46] The 400 MWt *FFTF* was a loop-type sodium-cooled, MOX-fueled fast reactor with no blanket for breeding additional plutonium. See figure 7.4.

Figure 7.4 **Fast Flux Test Facility at Hanford, Washington.**
Source: Federation of American Scientists.

Construction of the *FFTF* was completed in 1978 at the U.S. Department of Energy's (DOE) Hanford, Washington site, and criticality was achieved in 1980. It started serving as a test facility in 1982. When the CRBR was cancelled the following year, the *FFTF* lost its primary mission but continued to operate until April 1992 to test various aspects of fast reactor design and operation, including experiments designed to verify the ability to passively remove radioactive decay heat from a reactor core via convection of liquid-sodium coolant. By 1993, the usefulness of the reactor was diminishing, so the decision was taken in December of that

year to deactivate it. Over the next several years, efforts to find a new mission for *FFTF*, including producing radioactive isotopes for medical use or tritium for weapons, failed. With its fuel and sodium coolant removed, *FFTF* continues to be maintained in a cold standby condition, while proponents continue to seek new justifications for its use.

Alternative breeder concepts

Although the highest priority was given to LMFBRs, several other types of breeders were considered, and reached various stages of development in the United States. In addition to the LMFBR, these included the gas (helium) cooled fast breeder, and two thermal-neutron reactor types, the light-water breeder reactor and the molten-salt breeder reactor (MSBR). The fast-neutron breeder reactors were designed to breed plutonium from uranium-238, while the thermal-neutron breeder designs were optimized to breed uranium-233 from thorium-232.

Perhaps the most interesting alternate concept explored in this early work was the molten-salt breeder, which still has advocates.[47] In this reactor, the fuel and coolant are combined in a molten mixture of fluoride salts. The salt flows through the reactor core, through an intermediate heat exchanger, and then back to the reactor core. Molten-salt reactors were first proposed by Ed Bettis and Ray Briant of Oak Ridge National Laboratory (ORNL) during the post-World War II attempt to design a nuclear-powered aircraft.[48] Two molten-salt reactors were built at ORNL. The first was a prototype aircraft reactor, the 1.5 MWt *Aircraft Reactor Experiment (ARE)*, which operated for 100 hours in October 1954. The second, the graphite-moderated 8 MWt *Molten Salt Reactor Experiment (MSRE)*, operated between June 1965 and December 1969, demonstrating the technical feasibility of the molten-salt breeder concept.

In 1972, ORNL proposed a major development program that would have culminated in the construction and operation of a demonstration reactor called the Molten Salt Breeder Experiment. The total program cost was estimated at $350 million over a period of 11 years.[49] Those who would have had to approve the funding of the program were already heavily committed to the LMFBR, however. The ORNL proposal was rejected by the AEC partly because it wished to reduce the number of breeder candidates to be developed and because the breeding ratios projected for the molten-salt reactor were low compared to those foreseen for the fast-neutron reactors.[50] In January 1973, ORNL was directed to terminate MSBR development work. The program was reinstated a year later, and in 1974 ORNL submitted a more elaborate proposal calling for approximately $720 million to be spent over an 11-year period. This proposal was also rejected, and, in 1976, ORNL was again ordered to shut down the MSBR program "for budgetary reasons."[51]

The Shippingport Atomic Power Station was converted in 1975 into a marginal breeder using a thorium-uranium-233 fuel cycle.[52] The Shippingport plant had begun commercial operations on May 26, 1958 and was the first nuclear power

station in the United States to generate commercial electricity. It also was a major milestone in the development of light-water power reactors because it pioneered the use of uranium-oxide fuel in a water-cooled reactor.[53]

The gas-cooled, fast breeder reactor (GCFBR) was promoted by General Atomics, which had developed and was marketing the high temperature gas-cooled reactor (HTGR) in the United States.[54] The first HTGR demonstration plant was built at the Fort St. Vrain, Colorado Nuclear Generating Station. Fort St. Vrain was connected to the grid on December 11, 1976, and was shut down on August 29, 1989 due to continuing problems.[55] The GCFR would have had the same helium coolant technology, and its fuel would have had much in common with that of the HTGR. However, it would have lacked the graphite moderator of the HTGR and the safety advantage of its large thermal heat capacity.

AEC cost-benefit analyses

The AEC prepared three remarkably optimistic cost-benefit analyses of the LMFBR program. The first was written in 1968 and released in 1969;[56] the second was an updated (1970) analysis released in 1972,[57] and the third, a 1973 analysis, was first released as part of the AEC's 1974 Draft Environmental Impact Statement on the LMFBR Program.[58]

These analyses were extremely sensitive to changes in several important input variables, including the capital costs of LMFBRs relative to conventional nuclear reactors, electricity demand growth rates, uranium availability and the discount rate, which affects the relative weight given to near-term investments and long-term benefits. By making favorable but unrealistic assumptions, the AEC generated favorable benefit-to-cost ratios in each of these studies.

These assumptions included completely unrealistic nuclear power growth projections.[59] For example, figure 7.5 shows the 1974 AEC projections of nuclear power in which a total U.S. nuclear capacity of approximately 2000 gigawatt electric (GWe) was projected for 2008. 2000 GWe would have supplied approximately four times the U.S. actual total consumption of electricity in 2008. In reality, total U.S. nuclear capacity in 2008 was approximately 100 GWe and supplied approximately 20 percent of U.S. electrical power.

The rise and fall of Clinch River Demonstration Breeder Reactor

In 1969, statutory authorization was obtained to proceed with the first LMFBR demonstration plant,[60] financed in large part by the Federal Government.[61] The CRBR was to be a joint project of several electric utilities and the AEC (subsequently DOE).[62] The arrangements for financing, constructing, and managing the CRBR were spelled out in a 1972 Memorandum of Understanding and a subsequent series of detailed contracts among the AEC, Tennessee Valley Authority (TVA), Commonwealth Edison Co. (now Exelon), Project Management Corporation and
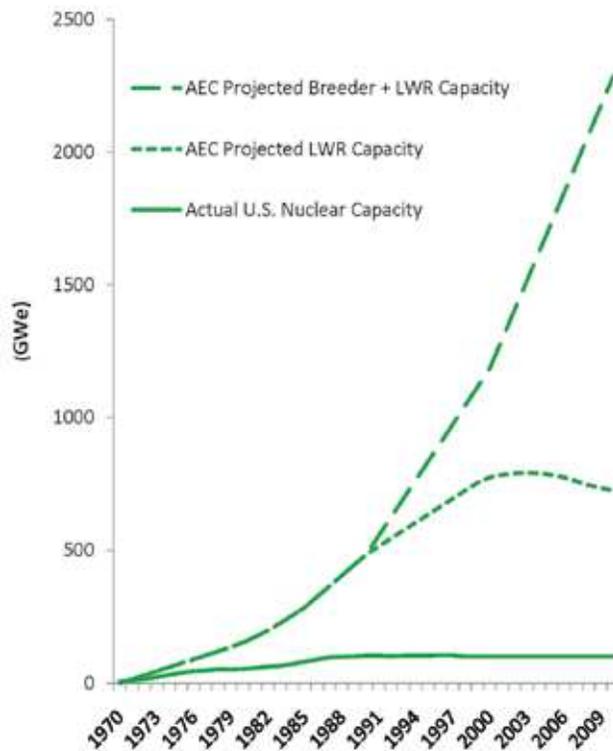
Figure 7.5 **AEC's 1974 estimate for the growth of nuclear power in the U.S.**
LWR represents light-water reactors. The AEC believed that U.S. uranium resources
could sustain less than 1000 GWe of light-water reactors. Source: U.S. Atomic Energy
Commission, Proposed Environmental Statement on the Liquid Metal Fast Breeder
Reactor WASH-1535 (1974).

the Breeder Reactor Corporation. Westinghouse Electric Corporation was selected
as the reactor manufacturer. Construction of the CRBR was projected to begin in
1974 or 1975 (and power generation in 1981 or 1982).

The plant was to be located at a bend in the Clinch River on the AEC site at Oak
Ridge, Tennessee, and to be operated by the TVA. It was to provide electricity to
the TVA grid. The CRBR was to be a bridge between the *FFTF* and an eventual
full-size prototype commercial breeder. Its design thermal power output was 975
MWt, approximately 2.5 times that of the *FFTF*, with an electrical generating
capacity of 350 MWe. The reactor was a loop-type sodium-cooled, MOX-fueled
plutonium breeder.

Starting in 1972, however, the LMFBR Program, and the CRBR project in particular began generating fierce public and political opposition due to economic, non-proliferation and safety concerns. On March 24, 1977, President Jimmy Carter, building on an October 28, 1976 decision by President Ford,[63] directed the indefinite deferral of commercial reprocessing and plutonium recycle in the United States. In the same directive, President Carter suspended the licensing process geared toward obtaining a Limited Work Authorization for the CRBR.[64]

The decisions by Presidents Ford and Carter were primarily in response to India's use of plutonium separated with U.S. assistance in an "Atoms for Peace" program to make a nuclear explosion in 1974. At the time, Brazil, Pakistan and South Korea had all contracted to buy reprocessing plants from France and Germany. The U.S. Government suspected that all three countries were interested in separating plutonium for weapons purpose.

Along with this concern about proliferation, the urgency of the breeder reactor began to fade. President Carter was advised that the AEC's projections of U.S. nuclear power growth and hence its claims that the United States would soon run out of low-cost uranium were greatly exaggerated.[65]

Cost increases also played a significant part in broadening opposition to the project. In September 1972, during hearings before the Joint Committee on Atomic Energy, the AEC presented a cost estimate of $699 million for the CRBR demonstration plant. The Federal Government would provide $422 million through the AEC and the utilities would provide the balance. The project was scheduled to achieve initial operation in 1979.[66] In the following year, the utilities committed themselves to pay $257 million plus interest, with a total utility commitment by September 1983 of $340 million. By the time detailed reference designs were completed in 1974, however, the estimated cost of the project had risen to $1.7 billion. By September 1983, approximately $1.7 billion had been spent and the estimated cost of the project had gone over $4 billion. According to the contract between the DOE and the utilities, virtually all of the additional funds would have had to be provided by the Government.[67]

A related issue was the high cost of building breeder reactors to produce electricity. Until late 1975, the AEC had been assuming that the capital costs of breeder reactors would decline to the same level as light-water reactors within 15 years. In 1977, this estimate was revised upward to a permanently higher cost of 25–75 percent. This meant that the cost of uranium would have to increase to $450–1350 per kg for the uranium savings to offset the additional capital charges of the breeder reactor.[68] Figure 1.2 in the Overview, chapter 1, shows the history of uranium prices since 1970.

In a study done for the conservative Heritage Foundation in 1982, Henry Sokolski, referring to contract studies done for the U.S. Arms Control and Disarmament Agency, noted that, given the assumed capital cost disparities, the breakeven price

for uranium would be nearly 18 times the then current price of uranium.[69] Such cost studies led many conservative groups to oppose the CRBR. The economics of breeder reactors appear as dim today as they did in 1983.[70]

Despite the Carter Administration's opposition, Congress continued to fund the CRBR. Although site construction could not proceed, the project continued to order and warehouse major components. In 1981, President Ronald Reagan restarted the process for licensing CRBR construction. By the end of 1982, the design was mostly complete and most components either were on hand or had been ordered.[71] But on October 23, 1983, Congress eliminated FY-1984 funding for the CRBR and, on December 15, 1983, the Nuclear Regulatory Commission terminated the licensing process and vacated the Limited Work Authorization it had granted the previous year. With this action, breeder reactor development in the United States essentially ended.

Efforts in the United States to resuscitate fast reactors

Since the cancellation of the CRBR in 1983, ANL and the Nuclear Energy program office in the DOE have continued to seek ways to revive fast-neutron reactor development in the United States, first by promoting the Integral Fast Reactor concept,[72] then through the Generation IV International Forum, and most recently the Global Nuclear Energy Partnership (GNEP).

Integral Fast Reactor and pyroprocessing

In the wake of the demise of the Clinch River Reactor project, ANL scientists developed and promoted the Integral Fast Reactor (IFR) concept. Patterned after the *EBR-II* with its Integral Fast Reactor fuel cycle facility (see *EBR-II* discussion), the IFR would integrate the plutonium-breeder reactor with an on-site spent fuel pyroprocessing and electro-refining process. In this process, plutonium and the minor transuranic elements would be separated and recycled together into new fuel.

The IFR was advanced as the key to making the breeder reactor economical, proliferation-resistant and environmentally acceptable.[73] There were ample grounds for skepticism, however. Most importantly, pyroprocessing looked still more expensive than conventional reprocessing. Moreover, were the IFR technology to be adopted by a non-weapon state it would provide the country with access to tons of plutonium in each co-located reactor and reprocessing facility. A cadre of experts trained in transuranic chemistry and plutonium metallurgy could separate out the plutonium from the other transuranic elements using hot cells and other facilities on-site. A 1992 study commissioned jointly by the U.S. Departments of Energy and State describes a variety of ways to use a pyroprocessing plant to produce relatively pure plutonium.[74]

Despite these problems, ANL was able to attract federal support for the IFR concept for a decade until the Clinton Administration cancelled the IFR program and the Congress terminated its funding in 1994. As a political compromise with Congress, it was agreed that while *EBR-II* would be shut down, funding of the fuel reprocessing research would continue—renaming it the "actinide recycling project."[75] A decade later this program would be re-characterized and promoted as necessary for long-term management of nuclear waste—becoming the centerpiece of the George W. Bush Administration's GNEP.

After Congress terminated funding for the IFR program, the DOE kept its pyroprocessing program alive by selecting it to process 3.35 metric tons of sodium-bonded *EBR-II* and *FFTF* spent fuel at INL. In 2006, the DOE estimated that pyroprocessing could treat the remaining 2.65 tons of this fuel in eight years at a cost of $234 million, including waste processing and disposal for a reprocessing cost of approximately $88,000/kg.[76]

International collaborations

R&D expenditures on advanced nuclear power reactors today are far less than in the 1970s (see figure 1.1, Overview, chapter 1). This has led to more international collaboration.

One such collaboration between government-funded nuclear R&D establishments is the Generation IV International Forum (Gen IV Forum). This forum was launched in 2001 at the instigation of the United States to facilitate international collaboration on the design of a new generation of nuclear reactors to be deployed after 2030. In 2002, the Forum selected six types for study, including three fast-neutron breeder reactors cooled respectively by liquid sodium, a liquid lead-bismuth alloy, and helium. Thus far, the collaborations on these efforts have focused on coordinating and pooling national research on reactor design, safety, proliferation resistance, fuel fabrication technologies, material development, and other topics.[77]

A second international collaboration, the International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) was initiated by a resolution of the International Atomic Energy Agency (IAEA) Board in 2001. In part because of the exclusion from the Gen IV Forum of Russia and other states with which the United States did not have agreements for nuclear cooperation. Thus far, INPRO has produced a report on "Guidance for the Evaluation of Innovative Nuclear Reactors and Fuel Cycles" and manuals on how to implement the assessment of "innovative nuclear-energy systems." Currently, INPRO members are collaborating on research projects and researchers from different countries are assessing proposed systems.[78]

In 2006, the George W. Bush Administration proposed GNEP with a goal of expanding nuclear power in the United States and abroad while reducing both the nuclear weapon proliferation risks and the requirements for long-term geological disposal of radioactive waste. To achieve these goals the Administration

proposed abandoning the once-through nuclear fuel, where nuclear fuel would be permanently sequestered in geologic repositories, in favor of the development and deployment of a closed fuel cycle based on advanced nuclear fuel reprocessing and fast-neutron "burner" reactors.

The GNEP program envisioned using fast-neutron reactors to burn rather than breed plutonium and the minor transuranic elements (neptunium, americium, and curium) to avoid having to place these long half-life radioactive materials into a geologic waste repository. The ratio of the number of fast reactors to conventional reactors depends upon the conversion ratio, defined as the ratio of the rate of production to the rate of destruction of the transuranic isotopes in the fast-neutron reactor. For fast-neutron reactors a wide range of conversion ratios is possible depending upon the reactor design. The lower the fast reactor conversion ratio, the fewer burner reactors would be required, with the number of fast burners proportional to $1/(1 - CR)$. In 1996, a National Research Council report cited General Electric as believing that the lowest possible conversion ratio that could be obtained using its PRISM fast reactor design, consistent with acceptable safety, as 0.6.[79] ANL more recently claims that a conversion ratio of 0.25 can be safely achieved.[80] Assuming the fast reactor conversion ratio is in the range of 0.25 to 0.6, 40–75 GWe of fast-reactor capacity would be required for every 100 GWe of light-water reactors.[81]

Despite the shift of mission from plutonium breeding to burning, the dream of breeding lives on. Although one ANL design of a fast-neutron burner reactor features a compact core where the inert (steel) blanket could not be readily converted to a blanket with uranium or depleted uranium, suitable for breeding, ANL in 2007 favored another design that could be converted to a breeder more easily but would cost more — on the order of 0.8 cents per kilowatt-hour.[82]

Conclusion

Although there are safety issues generic to liquid metal fast reactors, it does not appear that they were the predominant reasons for the demise of the breeder program in the United States. More important were proliferation concerns and a growing conviction that breeder reactors would not be needed or economically competitive with light-water reactors for decades, if ever.

Under GNEP, the DOE expressed renewed interest in fast reactors, initially as burner reactors to fission the actinides in the spent fuel of the light-water reactors. So far, the new designs are mostly paper studies, and the prospect of a strong effort to develop the burner reactors is at best uncertain. The Obama Administration has terminated the GNEP Programmatic Environmental Impact Statement and efforts by DOE to move to near-term commercialization of fast reactors and the closed fuel cycle for transmutation of waste. As this report went to press, it was debating whether to even continue R&D on fast-neutron reactors.[83] The economic and nonproliferation arguments against such reactors remain strong.

Endnotes

1 Much of the material in this and several of the following sections is based on Herbert Kouts, "The Development of Breeder Reactors in the United States," *Annual Review of Energy* 8 (1983): 383.

2 Catherine Westfall, "Vision and Reality: the *EBR-II* Story," *Nuclear News* 47 (2004): Part 2, 25.

3 Ibid.

4 Walter H. Zinn, a student of Fermi, joined Fermi's effort to build the first reactor, the "Chicago Pile," and became the first director of Argonne National Laboratory, which was created out of the Metallurgical Laboratory on July 1, 1946. By 1947, Zinn had directed construction of three experimental reactors; Richard G. Hewlett and Francis Duncan, *Atomic Shield: A History of the United States Atomic Energy Commission: Volume II 1947-1952* (University Park: Pennsylvania State University Press, 1969), 1972 reissue in paper of 1969 edition, 28.

5 Catherine Westfall, op. cit., 25–26.

6 Walter Mitchell, III and Stanley E. Turner, "Breeder Reactors," Oak Ridge: U.S. Atomic Energy Commission, Division of Technical Information (1971): 32.

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60 Public Law No. 91-44, 83 Stat. 46 (July 11, 1969) authorized the project definition phase of LMFBR program; Public Law No. 91-273, 84 Stat. 299 (June 2, 1970) authorized the AEC to enter into cooperative arrangement for construction of LMFBR demonstration plant; Public Law No. 92-84, 85 Stat. 304 (Aug. 11, 1971) increased appropriations for the demonstration plant program.

61 Public Law No. 91-273, op. cit. authorized expenditure of $50 million in cash, $20 million in services, and $10 million in waiver of charges for use of nuclear material for the demonstration plant. Public Law No. 92-84, op. cit., increased the cash commitment by an additional $50 million, bringing the total demonstration plant commitment to $130 million. House of Representative Report No. 92-325, op. cit., 24.

62 The Atomic Energy Commission was abolished on January 19, 1975, and replaced with two agencies: the Energy Research and Development Administration and the Nuclear Regulatory Commission. In October 1977, the U.S. Department of Energy was created to take over the Energy Research and Development Administration's functions plus energy functions from other federal agencies.

[63] "President Ford's Statement on Nuclear Policy," Presidential Documents, Vol. 12 no. 44 (October 28, 1976), 1624. "I have concluded that the reprocessing and recycling of plutonium should not proceed unless there is sound reason to conclude that the world community can effectively overcome the associated risks of proliferation."

[64] President Carter, "Nuclear Power Policy," (April 7 1977), Presidential Documents, Vol. 13: No. 15, 506. "(W)e will defer, indefinitely the commercial reprocessing and recycling of the plutonium produced in the U.S. nuclear power programs. From our own experience, we have concluded that a viable and economic nuclear power program can be sustained without such reprocessing and recycling".

[65] See, for example, T.B. Cochran, R.E. Train, F. von Hippel and R.H. Williams, "Proliferation Resistant Nuclear Power Technologies, Preferred Alternatives to the Plutonium Breeder," minority report of the Steering Committee of the LMFBR Program Review of the U.S. Energy Research and Development Administration (April 6, 1977), <http://docs.nrdc.org/nuclear/files/nuc_77042601a_21.pdf> (accessed 14 June, 2009).

[66] U.S. Government Accountability Office, "Report to Congress, The Liquid Metal Fast Breeder Reactor Program—Past, Present, and Future, op. cit., 26.

[67] Congressional Budget Office, "Comparative Analysis of Alternative Financing Plans for the Clinch River Breeder Reactor Project," Staff Working Paper, (September 20, 1983): 1–2.

[68] H. Feiveson, F. von Hippel, and R. Williams, "Fission Power: An Evolutionary Strategy," *Science* 203 (1979): 330. The range of $60–180 per pound U_3O_8 in 1976 dollars used in the article has been converted to 2006 dollars per kg of uranium using an inflation factor of approximately 2.9 and a factor of 2.59 pounds of U_3O_8 per kgU.

[69] Henry Sokolski, "The Clinch River Folly," The Heritage Foundation, *Backgrounder* 231 (December 3, 1982).

[70] Matthew Bunn, John Holdren, Steve Fetter, and Bob van der Zwaan, "The Economics of Reprocessing versus Direct Disposal of Spent Nuclear Fuel," *Nuclear Technology* 150 (2005). The authors show that direct disposal of spent light-water reactor fuel would be far less expensive than reprocessing and recycling in breeder reactors under a wide variety of assumptions. For example, for utility financing, central values for reprocessing and core fabrication of $1000 per kg of heavy metal and $1500 per kg heavy metal respectively, and a capital cost difference between light-water reactors and breeders of $200/KWe, the breakeven uranium price would be $340 per kg uranium, far greater than projected prices even if nuclear power grows substantially in the coming decades; see also, Erich Schneider and William Sailor, "Nuclear Fission," *Science & Global Security* 14 (2006): 183–211.

[71] Herbert Kouts, op. cit., 404.

72   This is the concept in which the spent fuel would be recycled onsite, Jack M. Holl, *Argonne National Laboratory, 1946–96* (Chicago: University of Illinois Press, 1997), 425, 426, 443–446.

73   Ibid.

74   R.G. Wymer et al., "An Assessment of the Proliferation Potential and International Implications of the Proliferation Potential and International Implications of the Integral Fast Reactor," Martin Marietta K/IPT-511 (May 1992); prepared for the Departments of State and Energy.

75   J. M. Holl, op. cit., 456.

76   U.S. Department of Energy, "Preferred Disposition Plan for Sodium-Bonded Spent Nuclear Reactor Fuel," Report to Congress (March 2006), tables 1 and 3. Pyroprocessing would account for 57 percent of the total cost, <http://www.ne.doe.gov/pdfFiles/DisPlanForSodBondedSNFMarch2006.pdf> (accessed 14 June 2009).

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82   Richard Garwin, "Plutonium Recycle in the U.S. Nuclear Power System," Federation of American Scientists, *Public Interest Report* 60 (2007): 11.

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Contributors

Thomas B. Cochran is a senior scientist in the nuclear program and holds the Wade Greene Chair for Nuclear Policy at the Natural Resources Defense Council (NRDC). He served as director of the nuclear program until 2007. He is a member of the Department of Energy's Nuclear Energy Advisory Committee. Cochran is the author of *The Liquid Metal Fast Breeder Reactor: An Environmental and Economic Critique* (Washington, D.C.: Resources for the Future, 1974). Cochran received his Ph.D. in physics from Vanderbilt University in 1967. He was a co-author of chapter 7, Fast Reactor Development in the United States.

Harold Feiveson is a Senior Research Scientist and Lecturer in Princeton University's Woodrow Wilson School. He has a PhD in public affairs from Princeton University (1972). Feiveson is the editor of *Science & Global Security*. Along with Professor von Hippel, he was the co-founder and co-director of the Program on Science and Global Security until July 2006. Feiveson was a co-author of Chapter 7, Fast Reactor Development in the United States.

Walt Patterson is Associate Fellow in the Energy, Environment and Development Programme at Chatham House in London, UK, and a Visiting Fellow at the University of Sussex. A postgraduate nuclear physicist, he has been actively involved in energy and environmental issues since the late 1960s. *Keeping The Lights On: Towards Sustainable Electricity* (Chatham House/Earthscan 2007, paperback 2009) is his thirteenth book. He has also published hundreds of papers, articles and reviews, on topics including nuclear power, coal technology, renewable energy systems, energy policy and electricity. He has been specialist advisor to two Select Committees of the House of Commons, an expert witness at many official hearings, a frequent broadcaster and advisor to media, and speaker or chair in conferences around the world. He has been awarded the Melchett Medal of the Energy Institute. The Scientific American 50 named him 'energy policy leader' for his advocacy of decentralized electricity. His current project for Chatham House and the Sussex Energy Group is called "Managing Energy: for climate and security". Patterson was the author of chapter 6, Fast Breeder Reactors in the United Kingdom.

Gennadi Pshakin is head of the Analytical Center for Nonproliferation at the Institute for Physics and Power Engineering (IPPE), Obninsk, and teaches at Obninsk Nuclear Technology University. Between 1985 and 1993, he worked as an IAEA safeguards inspector, and in 2003 was part of the IAEA team in Iraq. In the 1990s, he participated in negotiations on the trilateral initiative (USA – Russia – IAEA). Since 2001 he has been part of the INPRO project on developing a Proliferation Resistance Assessment Methodology and his recent research covers material protection, control, and accounting activities in Russia. His PhD (1980) was in nuclear engineering. Pshakin was the author of Chapter 5, The USSR-Russian Fast Neutron Reactor Program

M. V. Ramana is currently a Visiting Scholar with the Program in Science, Technology and Environmental Policy and the Program on Science and Global Security at the Woodrow Wilson School of Public and International Affairs, Princeton University. He has a PhD in physics (1994) and has held research positions at the University of Toronto, Massachusetts Institute of Technology, and Princeton University. He has taught at Boston University, Princeton University, and Yale University. His research focuses on India's nuclear energy and weapon programs. Currently, he is examining the economic viability and environmental impacts of the Indian nuclear power program. He is actively involved in the peace and anti-nuclear movements, and is associated with the Coalition for Nuclear Disarmament and Peace as well as Abolition-2000, a global network to abolish nuclear weapons. Ramana was the author of Chapter 3, India and Fast Breeder Reactors.

Mycle Schneider is an independent nuclear and energy consultant. He founded the Energy Information Agency WISE-Paris in 1983 and directed it until 2003. Since 1997 he has provided information and consulting services to the Belgian Energy Minister, the French and German Environment Ministries, the International Atomic Energy Agency, Greenpeace, the International Physicians for the Prevention of Nuclear War, the Worldwide Fund for Nature, the European Commission, the European Parliament's Scientific and Technological Option Assessment Panel and its General Directorate for Research, the Oxford Research Group, the French National Scientific Research Council, and the French Institute for Radiation Protection and Nuclear Safety. Since 2004 he has been in charge of the Environment and Energy Strategies lecture series for the International MSc in Project Management for Environmental and Energy Engineering Program at the French Ecole des Mines in Nantes. In 1997, along with Japan's Jinzaburo Takagi, he received Sweden's Right Livelihood Award "for serving to alert the world to the unparalleled dangers of plutonium to human life." Schneider was the author of chapter 2, Fast Breeder Reactors in France.

Tatsujiro Suzuki is an Associate Vice President of the Central Research Institute of Electric Power Industry, as well as a Senior Research Fellow at the Institute of Energy Economics of Japan. He is also a Visiting Professor at the Graduate School of Public Policy, University of Tokyo. He has a PhD in nuclear engineering from Tokyo University (1988). He was Associate Director of MIT's International Program on Enhanced Nuclear Power Safety from 1988–1993 and a Research Associate at MIT's Center for International Studies (1993–95) where he co-authored a report on Japan's plutonium program. For the past 20 years, he has been deeply involved in providing technical and policy assessments of the international implications of Japan's plutonium fuel-cycle policies and in examining the feasibility of interim spent-fuel storage as an alternative. He is a member of the Advisory Group on International Affairs of the Japan Atomic Energy Commission and now is also a member of the Ministry of Economy, Trade and Industry's Advisory Committee on Energy. Suzuki was the author of chapter 4, Japan's Plutonium Breeder Reactor and its Fuel Cycle.

Frank von Hippel is Professor of Public and International Affairs at Princeton University's Woodrow Wilson of Public and International Affairs. He has a PhD in nuclear physics (1962) from Oxford University. He is a co-founder of Princeton's Program on Science and Global Security. In the 1980s, as chairman of the Federation of American Scientists, he partnered with Evgenyi Velikhov in advising Mikhail Gorbachev on the technical basis for steps to end the nuclear arms race. In 1994–95, he served as Assistant Director for National Security in the White House Office of Science and Technology Policy. He has worked on fissile material policy issues for the past 30 years, including contributing to: ending the U.S. program to foster the commercialization of plutonium breeder reactors; convincing the United States and the Soviet Union to embrace the idea of a Fissile Material Production Cutoff Treaty; launching the U.S.-Russian cooperative nuclear materials protection, control and accounting program; and broadening efforts to eliminate the use of HEU in civilian reactors worldwide. Von Hippel was the author of chapter 1, Overview: The Rise and Fall of Plutonium Breeder Reactors and a co-author of chapter 7, Fast Reactor Development in the United States.