Small Nuclear Power Reactors
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*** There is revival of interest in small and simpler units for generating electricity from nuclear power, and for process heat.**
*** The interest is driven both by a desire to reduce capital costs and to provide power away from main grid systems.**
*** The technologies involved are very diverse.**

As nuclear power generation has become established since the 1950s, the size of reactor units has grown from 60 MWe to more than 1300 MWe, with corresponding economies of scale in operation. At the same time there have been many hundreds of smaller reactors built both for naval use (up to 190 MW thermal) and as neutron sources, yielding enormous expertise in the engineering of deliberately small units.

Today, due partly to the high capital cost of large power reactors generating electricity via the steam cycle and partly to consideration of public perception, there is a move to develop smaller units. These may be built independently or as modules in a larger complex, with capacity added incrementally as required. Economies of scale are provided by the numbers produced. There are also moves to develop small units for remote sites. The IAEA defines "small" as under 300 MWe. The most prominent modular project is the South African-led consortium developing the Pebble Bed Modular Reactor of of 170 MWe. Chinergy is preparing to build a similar unit, the 195 MWe HTR-PM in China. A US-led group is developing another design with 285 MWe modules. Both drive gas turbines directly, using helium as a coolant and operating at very high temperatures. They build on the experience of several innovative reactors in the 1960s and 1970s.
Generally, modern small reactors for power generation are expected to have greater simplicity of design, economy of mass production, and reduced siting costs. Many are also designed for a high level of passive or inherent safety in the event of malfunction (Traditional reactor safety systems are 'active' in the sense that they involve electrical or mechanical operation on command. Some engineered systems operate passively, eg pressure relief valves. Both require parallel redundant systems. Inherent or full Passive safety depends only on physical phenomena such as convection, gravity or resistance to high temperatures, not on functioning of engineered components).
Some are conceived for areas away from transmission grids and with small loads, others are designed to operate in clusters in competition with large units. The cost of electricity from a 50 MWe unit is estimated by DOE as 5.4 to 10.7 c/kWh (compared with charges in Alaska and Hawaii from 5.9 to 36.0 c/kWh).

US Congress is now funding research on both small modular nuclear power plants (assembled on site from factory-produced modules) and advanced gas-cooled designs (which are modular in the sense that up to ten or more units are progressively built to comprise a major power station). A US DOE report in 2001 considered nine designs which could possibly be deployed by 2010.
Already operating in a remote corner of Siberia are four small units at the Bilibino co-generation plant. These four 62 MWt (thermal) units are an unusual graphite-moderated boiling water design with water/steam channels through the moderator. They produce steam for district heating and 11 MWe (net) electricity each. They have performed well since 1976, much more cheaply than fossil fuel alternatives in the Arctic region.

Light Water Reactors
US experience has been of very small military power plants, such as the 11 MWt, 1.5 MWe (net) PM-3A reactor which operated at McMurdo Sound in Antarctica 1962-72, generating a total of 78 million kWh. There was also an Army program for small reactor development and some successful

small reactors from the main national program commenced in the 1950s. One was the Big Rock Point BWR of 67 MWe which operated for 35 years to 1997.
Of the following, the first three designs have conventional pressure vessel plus external steam generators (PV/loop design). The others mostly have the steam supply system inside the reactor pressure vessel ('integral' PWR design). All have enhanced safety features relative to current PWRs.
The Russian KLT-40S is a reactor well proven in icebreakers and now proposed for wider use in desalination and, on barges, for remote area power supply. Here a 150 MWt unit produces 35 MWe (gross) as well as up to 35 MW of heat for desalination or district heating (or 38.5 MWe gross if power only). These are designed to run 3-4 years between refuelling and it is envisaged that they will be operated in pairs to allow for outages (70% capacity factor), with on-board refuelling capability and spent fuel storage. At the end of a 12-year operating cycle the whole plant is taken to a central facility for overhaul and storage of spent fuel. Two units will be mounted on a 20,000 tonne barge.

Although the reactor core is normally cooled by forced circulation, the OKBM design relies on convection for emergency cooling. Fuel is uranium aluminium silicide with enrichment levels of up to 20%, giving up to 4-year refuelling intervals.
A larger Russian factory-built and barge-mounted unit (requiring a 12,000 tonne vessel) is the VBER-150, of 350 MW thermal, 110 MWe. It has modular construction and is derived by OKBM from naval designs, with two steam generators. Uranium oxide fuel enriched to 4.7% has burnable poison; it has low burnup (31 GWd/t average, 41.6 GWd/t max) and 8 year refuelling interval.
OKBM's larger VBER-300 PWR is a 325 MWe unit, the first of which will be built in Kazakhstan. It was originally envisaged in pairs as a floating nuclear power plant, displacing 49,000 tonnes. As a cogeneration plant it is rated at 200 MWe and 1900 GJ/hr. The reactor is designed for 60 year life and 90% capacity factor. It has four steam generators and a cassette core with 85 fuel assemblies enriched to 5% and 48 GWd/tU burn-up. Versions with three and two steam generators are also envisaged, of 230 and 150 MWe respectively. Also with more sophisticated and higher-enriched (18%) fuel in the core, the refuelling interval can be pushed from 2 years out to 15 years with burn-up to 125 GWd/tU. A 2006 joint venture between Atomstroyexport and Kazatomprom sets this up for development as a basic power source in Kazakhstan, then for export.
Another larger Russian reactor is the VK-300 boiling water reactor being developed for both power (250 MWe) and desalination (150 MWe plus 1675 GJ/hr). It has evolved from the VK-50 BWR at Dimitrovgrad, but uses standard components wherever possible, eg the reactor vessel of the VVER-1000. Fuel burn-up is 41 GWday/tU.

A smaller OKBM PWR unit under development is the ABV, with 45 MW thermal, 10-12 MWe output. It is compact, with integral steam generator and enhanced safety. The whole unit of some 600 tonnes will be factory-produced for ground or barge mounting - it would require a 2500 tonne barge. The core is similar to that of the KLT-40 except that enrichment is 16.5% and average burnup 95 GWd/t. Refuelling interval is about 8 years, and service life about 50 years
The CAREM (advanced small nuclear power plant) being developed by CNEA and INVAP in Argentina is a modular 100 MWt /27 MWe pressurised water reactor with integral steam generators designed to be used for electricity generation (27 MWe or up to 100 MWe) or as a research reactor or for water desalination (with 8 MWe in cogeneration configuration). CAREM has its entire primary coolant system within the reactor pressure vessel, self-pressurised and relying entirely on convection. Fuel is standard 3.4% enriched PWR fuel, with burnable poison, and is refuelled annually. It is a mature design which could be deployed within a decade.

On a larger scale South Korea's SMART (System-integrated Modular Advanced Reactor) is a 330 MWt pressurised water reactor with integral steam generators and advanced safety features. It is designed for generating electricity (up to 100 MWe) and/or thermal applications such as seawater

desalination. The design life is 60 years, with a 3-year refuelling cycle. A one-fifth scale plant (65 MWt) is being constructed, for operation in 2007.
The Japan Atomic Energy Research Institute (JAERI) is developing the MRX, a small (50-300 MWt) integral PWR reactor for marine propulsion or local energy supply (30 MWe). The entire plant would be factory-built. It has conventional 4.3% enriched PWR uranium oxide fuel with a 3.5-year refuelling interval and has a water-filled containment to enhance safety. It could be deployed within a decade.

Technicatome (Areva) in France has developed the NP-300 PWR from submarine power plants and aimed it at export markets for power, heat and desalination. It can be built for applications of 100 to 300 MWe or more with up to 500,000 m 3 /day desalination.
The Chinese NHR-200 is a simple and robust 200 MWt integral PWR used for district heating or desalination. It runs at lower temperature than the above designs. Spent fuel is stored around the core in the pressure vessel.
The International Reactor Innovative & Secure (IRIS) is being developed by Westinghouse as an advanced 3rd generation reactor. IRIS-50 is a modular 50 MWe or more pressurised water reactor with integral primary coolant system and circulation by convection. Fuel is similar to present LWRs. Enrichment is 5% with burnable poison and fuelling interval of 5 years (or longer with higher enrichment). IRIS-50 could be deployed this decade. (cf Advanced Reactors paper for larger IRIS.)

The Modular Simplified Boiling Water Reactor (MSBWR) is being developed by General Electric and Purdue University in USA at both 200 MWe and 50 MWe levels, based on GE's SBWR. It uses convection in the coolant and has 5% enriched BWR fuel with a 10-year refuelling interval. It may be ready for deployment this decade.
The TRIGA Power System is a PWR concept based on General Atomics' well-proven research reactor design. It is conceived as a 64 MWt, 16.4 MWe pool-type system operating at a relatively low temperature. The secondary coolant is organic perfluorocarbon. The fuel is uranium-zirconium hydride enriched to 20% and with a little burnable poison and requiring refuelling every 18 months. Spent fuel is stored inside the reactor vessel. (cf Research reactors paper).

Between 1966 and 1988, the AVR experimental pebble bed reactor at Juelich, Germany, operated for over 750 weeks at 15 MWe, most of the time with thorium-based fuel. The fuel consisted of about 100,000 billiard ball-sized fuel elements. The thorium was mixed with high-enriched uranium (HEU). Maximum burnups of 150 GWd/t were achieved. It was used to demonstrate the inherent safety of the design due to negative temperature coefficient: the helium coolant flow was cut off and the reactor power fell rapidly.
The 300 MWe THTR reactor in Germany was developed from the AVR and operated between 1983 and 1989 with 674,000 pebbles, over half containing Th/HEU fuel (the rest graphite moderator and some neutron absorbers). These were continuously recycled and on average the fuel passed six times through the core. Fuel fabrication was on an industrial scale. Several design features made the AVR unsuccessful, though the basic concept was again proven. It drove a steam turbine.
An 80 MWe HTR-modul was then designed by Siemens as a modular unit to be constructed in pairs. It was licensed in 1989, but was not constructed. This design was part of the technology bought by Eskom in 1996 and is a direct antecedent of PBMR

High-temperature Gas-cooled reactors
Building on the experience of several innovative reactors built in the 1960s and 1970s, new high-temperature gas-cooled reactors (HTRs) are being developed which will be capable of delivering high-temperature (up to 950°C) helium either for industrial application via heat exchanger or directly to drive gas turbines for electricity (the Brayton cycle) with almost 50% thermal efficiency possible (efficiency increases 1.5% with each 50°C increment). Technology developed in the last decade makes HTRs more practical than in the past, though the direct cycle means that there must be high integrity of fuel and reactor componentsFuel for these reactors is in the form of TRISO particles less than a millimetre in diameter. Each has a kernel (c0.5 mm) of uranium oxycarbide, with the uranium enriched up to 14% U-235. This is surrounded by layers of carbon and silicon carbide, giving a containment for fission products which is stable to 1600°C or more.
There are two ways in which these particles are arranged: in blocks - hexagonal 'prisms' of graphite, or in billiard ball-sized pebbles of graphite encased in silicon carbide, each with about 15,000 fuel particles and 9g uranium. Both have a high level of inherent safety, including strong negative temperature coefficient whereby fission slows as temperature rises. There is a greater amount of spent fuel than from the same capacity in a light water reactor.

The Japan Atomic Energy Research Institute's (JAERI) High-Temperature Test Reactor (HTTR) of 30 MW thermal started up at the end of 1998 and has been run successfully at 850°C. In 2004 it achieved 950°C outlet temperature. Its fuel is in 'prisms' and its main purpose is to develop thermochemical means of producing hydrogen from water.
Based on the HTTR, JAERI is developing the Gas Turbine High Temperature Reactor (GTHTR) of up to 600 MW thermal per module. It uses improved HTTR fuel elements with 14% enriched uranium achieving high burn-up (112 GWd/t). Helium at 850°C drives a horizontal turbine at 47% efficiency to produce up to 300 MWe. The core consists of 90 hexagonal fuel columns 8 metres high arranged in a ring, with reflectors. Each column consists of eight one-metre high elements 0.4 m across and holding 57 fuel pins made up of fuel particles with 0.55 mm diameter kernels and 0.14 mm buffer layer. In each 2-yearly refuelling, alternate layers of elements are replaced so that each remains for 4 years.
On the basis of four modules per plant, capital cost is projected at US$ 1300-1700/kWe and power cost about US 3.4 c/kWh.

China's HTR-10, a small high-temperature pebble-bed gas-cooled experimental reactor at the Institute of Nuclear & New Energy Technology (INET) at Tsinghua University north of Beijing started up in 2000 and reached full power in 2003. It has its fuel as a 'pebble bed' (27,000 elements) of oxide fuel with average burnup of 80 GWday/t U. Each pebble fuel element has 5g of uranium enriched to 17% in around 8300 particles. The reactor operates at 700°C (potentially 900°C) and has broad research purposes. Eventually it will be coupled to a gas turbine, but meanwhile it has been driving a steam turbine.
Construction of a larger version, the 200 MWe (450 MWt) HTR-PM, was approved in November 2005, with construction starting at the end of 2006. This will use 9% enriched fuel (520,000 elements) in an annular core giving 80 GWd/t discharge burnup. It will drive a steam cycle turbine. This Shidaowan demonstration reactor at Rongcheng in Shandong province is to pave the way for an 18-module full-scale power plant possibly at Weihei, also using the steam cycle. Plant life is envisaged as 60 years with 85% load factor.
China Huaneng Group, one of China's major generators, is the lead organization involved in the demonstration unit with 47.5% share; China Nuclear Engineering & Construction will have a 32.5% stake and Tsinghua University INET 20%. Projected cost is US$ 385 million (but later units falling to US$1500/kW with generating cost about 5c/kWh). Start-up is scheduled for 2010. The HTR-PM rationale is both eventually to replace conventional reactor technology for power, and also to provide for future hydrogen production.
In 2004 the small HTR-10 reactor was subject to an extreme test of its safety when the helium

circulator was deliberately shut off without the reactor being shut down. The temperature increased steadily, but the physics of the fuel meant that the reaction progressively diminished and eventually died away over three hours. At this stage a balance between decay heat in the core and heat dissipation through the steel reactor wall was achieved and the temperature never exceeded a safe 1600°C. This was one of six safety demonstration tests conducted then. The high surface area relative to volume, and the low power density in the core, will also be features of the full-scale units (which are nevertheless much smaller than most light-water types).

South Africa's **Pebble Bed Modular Reactor** (PBMR) is being developed by a consortium led by the utility Eskom, and drawing on German expertise. It aims for a step change in safety, economics and proliferation resistance. Production units will be 165 MWe. The PBMR will have a direct-cycle gas turbine generator and thermal efficiency about 41%, the helium coolant leaving the bottom of the core at about 900°C. Up to 450,000 fuel pebbles 60 mm diameter and containing 9g uranium enriched to 10% U-235 recycle through the reactor continuously (about six times each, taking six months) until they are expended, giving an average enrichment in the fuel load of 5% and average burn-up of 80 GWday/t U (eventual target burn-ups are 200 GWd/t). This means on-line refuelling as expended pebbles (which have yielded up to 91 GWd/t) are replaced, giving high capacity factor. The reactor core is lined with graphite and there is a central column of graphite as reflector. Control rods are in the side reflectors and cold shutdown units in the centre column.
Performance includes great flexibility in loads (40-100%) without loss of thermal efficiency, and with rapid change in power settings. Power density in the core is about one tenth of that in light water reactor, and if coolant circulation ceases the fuel will survive initial high temperatures while the reactor shuts itself down - giving inherent safety. Power control is by varying the coolant pressure and hence flow. Each unit will finally discharge about 35 tonnes/yr of spent pebbles to ventilated on-site storage bins.
Eventual construction cost (when in clusters of four or eight units) is expected to be very competitive. Investors in the PBMR project are Eskom, the South African Industrial Development Corporation and Westinghouse. A demonstration plant is due to be built in 2007 for operation in 2010. The first commercial units are expected on line in 2014 and Eskom has said it expects to order 24, which justify fully commercial fuel supply and maintenance. A contract for the pebble fuel plant at Pelindaba has been let.
Each 210g fuel pebble contains about 9g U and the total uranium in one fuel load is 4.1 t. MOX and thorium fuels are envisaged. With used fuel, the pebbles can be crushed and the 4% of their volume which is microspheres removed, allowing the graphite to be recycled. The company says microbial removal of C-14 is possible (also in the graphite reflectors when decommissioning).
A design certification application to the US Nuclear Regulatory Commission is expected in 2008, with approval expected in 2012, opening up world markets.

A larger US design, the **Gas Turbine - Modular Helium Reactor** (GT-MHR), will be built as modules of 285 MWe each directly driving a gas turbine at 48% thermal efficiency. The cylindrical core consists of 102 hexagonal fuel element columns of graphite blocks with channels for helium and control rods. Graphite reflector blocks are both inside and around the core. Half the core is replaced every 18 months. Burn-up is about 100 GWd/t, and coolant outlet temperature is 850°C with a target of 1000°C. It is being developed by General Atomics in partnership with Russia's OKBM, supported by Fuji (Japan) and Framatome ANP. Initially it will be used to burn pure ex-weapons plutonium at Seversk (Tomsk) in Russia. The preliminary design stage was completed in 2001. Plant costs are expected to be less than US$ 1000 /kW and total generating cost 2.9 c/kWh. The development timeline is for a prototype to be constructed in Russia 2006-09 following regulatory review there.
A smaller version of this, the Remote-Site Modular Helium Reactor (RS-MHR) of 10-25 MWe has been proposed by General Atomics. The fuel would be 20% enriched and refuelling interval would be 6-8 years.

A third full-size HTR design is Areva's Very High Temperature Reactor (VHTR) being put forward by Framatome ANP. It is based on the GT-MHR and has also involved Fuji. Reference design is 600 MW (thermal) with prismatic block fuel like the GT-MHR. Target core outlet temperature is 1000°C and it uses and indirect cycle, possibly with a helium-nitrogen mix in the secondary system. This removes the possibility of contaminating the generation or hydrogen production plant with radionuclides from the reactor core.

HTRs can potentially use thorium-based fuels, such as HEU with Th, U-233 with Th, and Pu with Th. Most of the experience with thorium fuels has been in HTRs.

The three larger HTR designs, with the AHTR described below, are contenders for the US Next-Generation Nuclear Plant.

A small US HTR concept is the Adams Atomic Engines 10 MWe direct simple cycle plant with nitrogen as the reactor coolant and working fluid. The reactor core will be a fixed, annular bed with about 80,000 fuel elements each 6 cm diameter and containing approximately 9 grams of heavy metal with expected average burn-up of 80 GWd/t. The initial units will provide a reactor core outlet temperature of 800°C and a thermal efficiency near 25%. A demonstration plant is proposed for completion by 2011 with series production by 2014.

Liquid Metal cooled Fast Reactors

Fast neutron reactors have no moderator, a higher neutron flux and are normally cooled by liquid metal such as sodium, lead, or lead-bismuth, with high conductivity and boiling point. They operate at or near atmospheric pressure and have passive safety features (most have convection circulating the primary coolant). Automatic load following is achieved due to the reactivity feedback - constrained coolant flow leads to higher core temperature which slows the reaction. Primary coolant flow is by convection. They typically use boron carbide control rods.

The Encapsulated Nuclear Heat Source (ENHS) is a liquid metal-cooled reactor concept of 50 MWe being developed by the University of California. The core is at the bottom of a metal-filled module sitting in a large pool of secondary molten metal coolant which also accommodates the 8 separate and unconnected steam generators. There is convection circulation of primary coolant within the module and of secondary coolant outside it. Outside the secondary pool the plant is air cooled. Control rods would need to be adjusted every year or so and load-following would be autonomous. The whole reactor sits in a 17 metre deep silo. Fuel is a uranium-zirconium alloy with 13% U enrichment (or U-Pu-Zr with 11% Pu) with a 15-20 year life. After this the module is removed, stored on site until the primary lead (or Pb-Bi) coolant solidifies, and it would then be shipped as a self-contained and shielded item. A new fuelled module would be supplied complete with primary coolant. The ENHS is designed for developing countries and is highly proliferation-resistant but is not yet close to commercialisation.

A related project is the Secure Transportable Autonomous Reactor - STAR being developed by Argonne under the leadership of Lawrence Livermore Laboratory (DOE). It a lead-cooled fast neutron modular reactor with passive safety features. Its 400 MWt. size means it can be shipped by rail and cooled by natural circulation. It uses U-transuranic nitride fuel in a cassette which is replaced every 15-20 years. The STAR-LM was conceived for power generation, running at 578°C and producing 180 MWe.

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

A smaller STAR variant is the Small Sealed Transportable Autonomous Reactor - SSTAR, being developed in collaboration with Toshiba and others in Japan (see 4S below). It has lead or Pb-Bi cooling, runs at 566°C and has integral steam generator inside the sealed unit, which would be installed below ground level. Conceived in sizes 10-100 MWe, main development is now focused on a 45 MWt/ 20 MWe version as part of the US Generation IV effort. After a 20-year life without

refuelling, the whole reactor unit is then returned for recycling the fuel. The core is one metre diameter and 0.8m high. SSTAR will eventually be coupled to a Brayton cycle turbine using supercritical carbon dioxide. Prototype envisaged 2015.
For all STAR concepts, regional fuel cycle support centres would handle fuel supply and reprocessing, and fresh fuel would be spiked with fission products to deter misuse. Complete burnup of uranium and transuranics is envisaged in STAR-H2, with only fission products being waste.
Japan's LSPR is a lead-bismuth cooled reactor of 150 MWt /53 MWe. Fuelled units would be supplied from a factory and operate for 30 years, then be returned. Concept intended for developing countries.
A small-scale design developed by Toshiba Corporation in cooperation with Japan's Central Research Institute of Electric Power Industry (CRIEPI) and funded by the Japan Atomic Energy Research Institute (JAERI) is the 5 MWt, 200 kWe Rapid-L, using lithium-6 (a liquid neutron poison) as control medium. It would have 2700 fuel pins of 40-50% enriched uranium nitride with 2600°C melting point integrated into a disposable cartridge. The reactivity control system is passive, using lithium expansion modules (LEM) which give burnup compensation, partial load operation as well as negative reactivity feedback. As the reactor temperature rises, the lithium expands into the core, displacing an inert gas. Other kinds of lithium modules, also integrated into the fuel cartridge, shut down and start up the reactor. Cooling is by molten sodium, and with the LEM control system, reactor power is proportional to primary coolant flow rate. Refuelling would be every 10 years in an inert gas environment. Operation would require no skill, due to the inherent safety design features. The whole plant would be about 6.5 metres high and 2 metres diameter.
The Super-Safe, Small & Simple - 4S 'nuclear battery' system is being developed by Toshiba and CRIEPI in Japan in collaboration with STAR work in USA. It uses sodium as coolant (with electromagnetic pumps) and has passive safety features, notably negative temperature and void reactivity. The whole unit would be factory-built, transported to site, installed below ground level, and would drive a steam cycle. It is capable of three decades of continuous operation without refuelling. Metallic fuel (169 pins 10mm diameter) is uranium-zirconium enriched to less than 20% or U-Pu-Zr alloy with 24% Pu for the 10 MWe version or 11.5% Pu for the 50 MWe version. Steady power output over the core lifetime is achieved by progressively moving upwards an annular reflector around the slender core (0.68m diameter, 2m high in the 10 MWe version, 1.2m diameter and 2.5m high in the 50 MWe version). After 14 years a neutron absorber at the centre of the core is removed and the reflector repeats its slow movement up the core for 16 more years. In the event of power loss the reflector falls to the bottom of the reactor vessel, slowing the reaction, and external air circulation gives decay heat removal.
Both 10 MWe and 50 MWe versions of 4S are designed to automatically maintain an outlet coolant temperature of 510¡C - suitable for power generation with high temperature electrolytic hydrogen production. Plant cost is projected at US$ 2500/kW and power cost 5-7 cents/kWh for the small unit - very competitive with diesel in many locations. The design has gained considerable support in Alaska and toward the end of 2004 the town of Galena granted initial approval for Toshiba to build a 4S reactor in that remote location. A pre-application NRC review is being sought with a view to a demonstration unit operating by 2012. Its design is sufficiently similar to PRISM - GE's modular 150 MWe liquid metal-cooled inherently-safe reactor which went part-way through US NRC approval process for it to have good prospects of licensing.
The L-4S is Pb-Bi cooled version of 4S.
A significant fast reactor prototype was the EBR-II, a fuel recycle reactor of 62 MWt at Argonne which used the pyrometallurgically-refined spent fuel from light water reactors as fuel, including a wide range of actinides. The objective of the program is to use the full energy potential of uranium rather then only about one percent of it. It is shut down and being decommissioned. An EBR-III of 200-300 MWe was proposed but not developed.
Russia has experimented with several lead-cooled reactor designs, and has used lead-bismuth cooling for 40 years in its submarine reactors. Pb-208 (54% of naturally-occurring lead) is

transparent to neutrons. A significant Russian design is the BREST fast neutron reactor, of 300 MWe or more with lead as the primary coolant, at 540°C, and supercritical steam generators. The core sits in a pool of lead at near atmospheric pressure. It is inherently safe and uses a U+Pu nitride fuel. No weapons-grade Pu can be produced (since there is no uranium blanket), and spent fuel can be recycled indefinitely, with on-site facilities. A pilot unit is being built at Beloyarsk and 1200 MWe units are planned.

A smaller and newer Russian design is the Lead-Bismuth Fast Reactor (SVBR) of 75-100 MWe. This is an integral design, with the steam generators sitting in the same Pb-Bi pool at 400-480°C as the reactor core, which could use a wide variety of fuels. The unit would be factory-made and shipped as a 4.5m diameter, 7.5m high module, then installed in a tank of water which gives passive heat removal and shielding. A power station with 16 such modules is expected to supply electricity at lower cost than any other new Russian technology as well as achieving inherent safety and high proliferation resistance. (Russia built 7 Alfa-class submarines, each powered by a compact 155 MWt Pb-Bi cooled reactor, and 70 reactor-years operational experience was acquired with these.)

Molten Salt Reactors

During the 1960s the USA developed the molten salt breeder reactor as the primary back-up option for the fast breeder reactor (cooled by liquid metal) and a small prototype was operated at Oak Ridge. There is now renewed interest in the concept in Japan, Russia, France and the USA.

In the Molten Salt Reactor (MSR) the fuel is a molten mixture of lithium and beryllium fluoride salts with dissolved thorium and U-233 fluorides. The core consists of unclad graphite moderator arranged to allow the flow of salt at some 700°C and at low pressure. Heat is transferred to a secondary salt circuit and thence to steam. The fission products dissolve in the salt and are removed continuously in an on-line reprocessing loop and replaced with Th-232 or U-238. Actinides remain in the reactor until they fission or are converted to higher actinides which do so.

The FUJI MSR is a 100 MWe design operating as a near-breeder and being developed internationally by a Japanese, Russian and US consortium.

The attractive features of this MSR fuel cycle include: the high-level waste comprising fission products only, hence shorter-lived radioactivity; small inventory of weapons-fissile material (Pu-242 being the dominant Pu isotope); low fuel use (the French self-breeding variant claims 50kg of thorium and 50kg U-238 per billion kWh); and safety due to passive cooling up to any size.

The Advanced High-temperature Reactor (AHTR) is a larger reactor using a coated-particle graphite-matrix fuel like that in the GTMHR (see above section) and with molten fluoride salt as primary coolant. While similar to the gas-cooled HTR it operates at low pressure (less than 1 atmosphere) and higher temperature, and gives better heat transfer than helium. The salt is used solely as coolant, and achieves temperatures of 750-1000°C while at low pressure. This could be used in thermochemical hydrogen manufacture. Reactor sizes of 1000 MWe/2400 MWt are envisaged, with capital costs estimated at less than $1000/kW.

Molten fluoride salts are a preferred interface fluid between the nuclear heat source and any chemical plant. The aluminium smelting industry provides substantial experience in managing them safely. The hot molten salt can also be used with secondary helium coolant generating power via the Brayton cycle.

Primary coolants

The advent of some of the designs mentioned above provides opportunity to review the various primary coolants used in nuclear reactors:

Water or heavy water must be maintained at very high pressure (1000-2200 psi, 7-15 MPa) to enable it to function above 100°C, as in present reactors. This has a major influence on reactor engineering. However, supercritical water around 25 MPa can give 45% thermal efficiency - as at some fossil-fuel power plants today with outlet temperatures of 600°C, and at ultra supercritical levels (30+ MPa) 50% may be attained.

Helium must be used at similar pressure (1000-2000 psi, 7-14 MPa) to maintain sufficient density for efficient operation. Again, there are engineering implications, but it can be used in the Brayton cycle to drive a turbine directly.
Carbon dioxide was used in early British reactors and their AGRs. It is denser than helium and thus likely to give better thermal conversion efficiency. There is now interest in supercritical CO_2 for the Brayton cycle.
Sodium, as normally used in fast neutron reactors, melts at 98°C and boils at 883°C at atmospheric pressure, so despite the need to keep it dry the engineering required to contain it is relatively modest. However, normally water/steam is used in the secondary circuit to drive a turbine (Rankine cycle) at lower thermal efficiency than the Brayton cycle.
Lead or lead-bismuth are capable of higher temperature operation. They are transparent to neutrons, aiding efficiency, and do not react with water. However, they are corrosive of fuel cladding and steels, and Pb-Bi yields Po activation products. Pb-Bi melts at 125°C and boils at 1670°C, Pb melts at 327°C and boils at 1737°C. In 1998 Russia declassified a lot of research information derived from its experience with submarine reactors, and US interest in using Pb/Pb-Bi for small reactors has increased subsequently.
Molten fluoride salt boils at 1400°C at atmospheric pressure, so allows several options for use of the heat, including using helium in a secondary Brayton cycle with thermal efficiencies of 48% at 750°C to 59% at 1000°C, or manufacture of hydrogen.

Low-pressure liquid coolants allow all their heat to be delivered at high temperatures, since the temperature drop in heat exchangers is less than with gas coolants. Also, with a good margin between operating and boiling temperatures, passive cooling for decay heat is readily achieved.
Other, mostly larger, designs are described in the Advanced Reactors Paper.

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