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.**
- **This interest in small and medium nuclear power reactors is driven both by a desire to reduce capital costs and to provide power away from large 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 1600 MWe, with corresponding economies of scale in operation. At the same time there have been many hundreds of smaller power reactors built both for naval use (up to 190 MW thermal) and as neutron sources[a], yielding enormous expertise in the engineering of small units. The International Atomic Energy Agency (IAEA) defines 'small' as under 300 MWe, and up to 700 MWe as 'medium' – including many operational units from 20th century. Together they are now referred to as small and medium reactors (SMRs). This paper focuses on advanced designs in the small category, *i.e.* those now being built for the first time or still on the drawing board

Today, due partly to the high capital cost of large power reactors generating electricity via the steam cycle and partly to the need to service small electricity grids under about 4 GWe,[b] 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 (see section below on Modular construction using small reactor units). Economies of scale are provided by the numbers produced. There are also moves to develop small units for remote sites.

Generally, modern small reactors for power generation are expected to have greater simplicity of design, economy of mass production, and reduced siting costs. Most are also designed for a high level of passive or inherent safety in the event of malfunction[c]. A 2010 report by a special committee convened by the American Nuclear Society showed that many safety provisions necessary, or at least prudent, in large reactors are not necessary in the small designs forthcoming[d].

A 2009 assessment by the IAEA under its Innovative Nuclear Power Reactors & Fuel Cycle (INPRO) program concluded that there could be 96 small modular reactors (SMRs) in operation around the world by 2030 in its 'high' case, and 43 units in the 'low' case, none of them in the USA.

The most advanced modular project is in China, where Chinergy is preparing to build the 210 MWe HTR-PM, which consists of twin 250 MWt reactors. In South Africa, Pebble Bed Modular Reactor (Pty) Limited and Eskom had been developing the pebble bed modular reactor (PBMR) of 200 MWt (80 MWe), but funding for this project has been stopped. A US group led by General Atomics is developing another design – the gas turbine modular helium reactor (GT-MHR) – with 600 MWt (285 MWe) modules driving a gas turbine directly, using helium as a coolant and operating at very high temperatures. All three are high-temperature gas-cooled reactors (HTRs) which build on the experience of several innovative reactors in the 1960s and 1970s.

Another significant line of development is in very small fast reactors of under 50 MWe. 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.

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.

Also in the small reactor category are the Indian 220 MWe pressurised heavy water reactors (PHWRs) based on Canadian technology, and the Chinese 300-325 MWe PWR such as built at Qinshan Phase I and at Chashma in Pakistan, and now called CNP-300. These designs are not detailed in this paper simply because they are well-established. The Nuclear Power Corporation of India (NPCIL) is now focusing on 540 MWe and 700 MWe versions of its PHWR, and is offering both 220 and 540 MWe versions internationally. These small established designs are relevant to situations requiring small to medium units, though they are not state of the art technology.

Other, mostly larger new designs are described in the information page on Advanced Nuclear Power Reactors.

Medium and Small (25 MWe up) reactors with development well advanced

Name	Capacity	Type	Developer
KLT-40S	35 MWe	PWR	OKBM, Russia
VK-300	300 MWe	PWR	Atomenergoproekt, Russia
CAREM	27 MWe	PWR	CNEA & INVAP, Argentina
IRIS	100-335 MWe	PWR	Westinghouse-led, international
mPower	125 MWe	PWR	Babcock & Wilcox, USA
SMART	100 MWe	PWR	KAERI, South Korea
NuScale	45 MWe	PWR	NuScale Power, USA
HTR-PM	2x105 MWe	HTR	INET & Huaneng, China
PBMR	80 MWe	HTR	Eskom, South Africa
GT-MHR	285 MWe	HTR	General Atomics (USA), Rosatom (Russia)
BREST	300 MWe	LMR	RDIPE, Russia
SVBR-100	100 MWe	LMR	Rosatom/En+, Russia
FUJI	100 MWe	MSR	ITHMSO, Japan-Russia-USA

Light water reactors

US experience of small light water reactors (LWRs) 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 designs, the KLT and VBER have conventional pressure vessels 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 LWRs.

KLT-40S

Russia's KLT-40S from OKBM Afrikantov 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 with on-board refuelling capability and used 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 used fuel. Two units will be mounted on a 20,000 tonne barge to allow for outages (70% capacity factor).

Although the reactor core is normally cooled by forced circulation, the design relies on convection for emergency cooling. Fuel is uranium aluminium silicide with enrichment levels of up to 20%, giving up to four-year refuelling intervals.

The first floating nuclear power plant, the *Akademik Lomonosov*, commenced construction in 2007 and is planned to be located near to Vilyuchinsk. The plant is due to be completed in 2011.[2]

RITM-200

OKBM Afrikantov is developing a new icebreaker reactor – RITM-200 – to replace the KLT reactors and to serve in floating nuclear power plants. This is an integral 210 MWt, 55 MWe PWR with inherent safety features. A single compact RITM-200 could replace twin KLT-40S (but yielding less total power). A major challenge is the reliability of steam generators and associated equipment which are much less accessible when inside the reactor pressure vessel.

VBER-150, VBER-300

A larger Russian factory-built and barge-mounted unit (requiring a 12,000 tonne vessel) is the VBER-150, of 350 MWt, 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 burn-up (31 GWd/t average, 41.6 GWd/t maximum) and eight-year refuelling interval.

OKBM Afrikantov's larger VBER-300 PWR is a 295 MWe unit, the first of which is planned to 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 two 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[e].

VK-300

Another larger Russian reactor is the VK-300 boiling water reactor being developed specifically for cogeneration of both power and district heating or heat for desalination (150 MWe plus 1675 GJ/hr) by the N.A. Dollezhal Research and Development Institute of Power Engineering (NIKIET). It has evolved from the 50 MWe (net) VK-50 BWR at Dimitrovgrad[f], but uses standard components wherever possible, and fuel elements similar to the VVER. Cooling is passive, by convection, and all safety systems are passive. Fuel burn-up is 41 GWd/tU. It is capable of producing 250 MWe if solely electrical. In September 2007 it was announced that six would be built at Kola and at Primorskaya in the far east, to start operating 2017-20.[3]

VKT-12

A smaller Russian BWR design is the 12 MWe transportable VKT-12, described as similar to the VK-50 prototype BWR at Dimitrovgrad, with one loop. It has a ceramic-metal core with uranium enriched to 2.4-4.8%, and 10-year refuelling interval. The reactor vessel is 2.4m inside diameter and 4.9 m high.

ABV

A smaller Russian OKBM Afrikantov PWR unit under development is the ABV, with a range of sizes from 45 MWt (ABV-6M) down to 18 MWt (ABV-3), giving 4-18 MWe outputs. The units are compact, with integral steam generator. The whole unit will be factory-produced for ground or barge mounting – the ABV-6M would require a 3500 tonne barge; the ABV-3, 1600 tonne. The core is similar to that of the KLT-40 except that enrichment is 16.5% and average burn-up 95 GWd/t. Refuelling interval is about 8-10 years, and service life about 50 years.

CAREM

The CAREM reactor being developed by INVAP in Argentina[g], under contract to the Argentine National Atomic Energy Commission (CNEA), is a modular 100 MWt (27 MWe) pressurised water reactor with integral steam generators designed to be used for electricity generation 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, and scaled up to 300 MWe or more. The prototype is to be built in the northwestern Formosa province of Argentina[4].

SMART

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 by the Korea Atomic Energy Research Institute (KAERI) for generating electricity (up to 100 MWe) and/or thermal applications such as seawater desalination. Design life is 60 years, with a three-year refuelling cycle. While the basic design is complete, the absence of any orders for an initial reference unit has stalled development. KAERI is now intending to proceed to licensing the design by 2012.

MRX

The Japan Atomic Energy Research Institute (JAERI) designed 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. Little has been heard of it since the start of the Millennium.

NP-300

Technicatome (Areva TA) in France has developed the NP-300 PWR from submarine power plants and aimed it at export markets for power, heat and desalination. It has passive safety systems and can be built for applications of 100 to 300 MWe or more with up to 500,000 m^3/day desalination. Areva TA makes the K15 naval reactor of 150 MW, running on low-enriched fuel, and the land-based equivalent: *Réacteur d'essais à terre* (RES) a test version of which is under construction at Cadarache.

NHR-200

The Chinese NHR-200 (Nuclear Heating Reactor), developed by Tsingua University's Institute of Nuclear Energy Technology (now the Institute of Nuclear and New Energy Technology), is a simple 200 MWt integral PWR design for district heating or desalination. It is based on the NHR-5 which was commissioned in 1989, and runs at lower temperature than the above designs[h]. Used fuel is stored around the core in the pressure vessel. In 2008, the Chinese government was reported to have agreed to build a multi-effect distillation (MED) desalination plant using this on the Shandong peninsula.

IRIS

Westinghouse's IRIS (International Reactor Innovative & Secure) is an advanced 3rd generation reactor. A 335 MWe capacity is proposed, although it could be scaled down to around 100 MWe. IRIS is a modular pressurised water reactor with integral primary coolant system and circulation by convection. Fuel is similar to present LWRs and (at least for the 335 MWe version) fuel assemblies are identical to those in AP1000, according to Westinghouse. Enrichment is 5% with burnable poison and fuelling interval of four years (or longer with higher enrichment). US design certification is at the pre-application state.

mPower

In mid-2009, Babcock & Wilcox (B&W) announced its B&W mPower reactor, a 125 MWe integral PWR designed to be factory-made and railed to site[i]. The reactor pressure vessel containing core of 2x2 metres and steam generator is thus only 3.6 metres diameter and 22 m high, and the whole unit 4.5 m diameter and 23 m high. It would be installed below ground, have an air-cooled condenser giving 31% thermal efficiency, and passive safety systems. With cold water source for condensers the efficiency increases and capacity is up to 136 MWe. The integral steam generator is derived from naval designs, as is the control rod set-up. It has a "conventional core and standard fuel"[j] (< 20 t) enriched to 5%, with burnable poisons, to give a five-year operating cycle between refuelling, which will involve replacing the entire core as a single cartridge. Burn-up is less than 40 GWd/t. (B&W draws upon over 50 years experience in manufacturing nuclear propulsion systems for the US Navy, involving compact reactors with long core life.) A 60-year service life is envisaged, as sufficient used fuel storage would be built on site for this.

The mPower reactor is modular in the sense that several units would be combined into a power station of any size, but most likely 500-750 MWe and using 250 MWe turbine generators (also shipped as complete modules), constructed in three years. B&W's present manufacturing capability in North America can produce these units, and it has set up B&W Modular Nuclear Energy LLC to market the design. The company intends to apply for design certification in mid-2012, with a view to a combined construction and operating licence application in 2013, construction start in 2015 and operation of the first unit in 2018.

When B&W announced the launch the mPower design, it said that Tennessee Valley Authority (TVA) would begin the process of evaluating Clinch River at Oak Ridge as a potential lead site for the mPower reactor, and that a Memorandum of Understanding has been signed by B&W, TVA and a consortium of regional municipal and cooperative utilities to explore the construction of a fleet of mPower reactors. It was later reported that the other signatories of the agreement are First Energy and Oglethorpe Power[5].

NuScale

A smaller unit is the NuScale multi-application small PWR, a 160 MWt or 45 MWe integral PWR which is apparently similar to IRIS but with natural circulation. It will be factory-built with 3 metre diameter pressure vessel and convection cooling, with the only moving parts being the control rod drives. It uses standard PWR fuel enriched to < 4.95% in normal PWR fuel assemblies (but which are only 1.8 m long), with 24-month refuelling cycle. Installed in a water-filled pool below ground, the 4.3 m diameter, 18 m high cylindrical containment vessel module weighs 450 tonnes and contains the reactor and steam generator. A standard power plant would have 12 modules together giving about 500 MWe. An overhead crane would hoist each module from its pool to a separate part of the plant for refueling.

An application for US design certification is expected early in 2012 and there are hopes for a first operating unit in 2018. The NuScale Power company was spun out of Oregon Sate University in 2007, though the technology originates in the US Department of Energy. The company estimates in 2010 that overnight capital cost for a 12-module, 540 MWe NuScale plant is about $4000 per kilowatt.

TRIGA

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 perfluorocarbon. The fuel is uranium-zirconium hydride enriched to 20% and with a little burnable poison and requiring refuelling every 18 months. Used fuel is stored inside the reactor vessel.

High-temperature gas-cooled reactors

Building on the experience of several innovative reactors built in the 1960s and 1970s[k], new high-temperature gas-cooled reactors (HTRs) are being developed which will be capable of delivering high temperature (up to about 1000°C) helium either for industrial application via a heat exchanger, or to make steam conventionally via a steam generator, or directly to drive a Brayton cycle gas turbine for electricity with almost 50% thermal efficiency possible (efficiency increases around 1.5% with each 50°C increment). Improved metallurgy and 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 components. All but one of those described below have neutron moderation, one is a fast neutron reactor.

Fuel for these reactors is in the form of TRISO (tristructural-isotropic) particles less than a millimetre in diameter. Each has a kernel (*ca.* 0.5 mm) of uranium oxycarbide (or uranium dioxide), with the uranium enriched up to 20% U-235, though normally less. This is surrounded by layers of carbon and silicon carbide, giving a containment for fission products which is stable to over 1600°C.

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. There is a greater amount of used fuel than from the same capacity in a light water reactor. The moderator is graphite.

HTRs can potentially use thorium-based fuels, such as highly-enriched or low-enriched uranium with Th, U-233 with Th, and Pu with Th. Most of the experience with thorium fuels has been in HTRs (see information paper on Thorium).

With negative temperature coefficient of reactivity (the fission reaction slows as temperature increases) and passive decay heat removal, the reactors are inherently safe. HTRs therefore do not require any containment building for safety. They are sufficiently small to allow factory fabrication, and will usually be installed below ground level.

Three HTR designs in particular – PBMR, GT-MHR and Antares – are contenders for the Next Generation Nuclear Plant (NGNP) project in the USA (see Next Generation Nuclear Plant section in the information page on US Nuclear Power Policy).

HTTR, GTHTR

Japan Atomic Energy Research Institute's (JAERI's) High-Temperature Test Reactor (HTTR) of 30 MWt started up at the end of 1998 and has been run successfully at 850°C for 30 days. 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 MWt 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 two-yearly refuelling, alternate layers of elements are replaced so that each remains for four years.

HTR-10

China's HTR-10, a 10 MWt high-temperature 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 burn-up of 80 GWday/t U. Each pebble fuel element has 5g of uranium enriched to 17% in around 8300 TRISO-coated 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.

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, the temperature never exceeded a safe 1600°C, and there was no fuel failure. 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).

HTR-PM

Construction of a larger version of the HTR-10, China's HTR-PM, was approved in principle in November 2005, with construction starting in 2010. This was to be a single 200 MWe (450 MWt) unit but will now have twin reactors, each of 250 MWt driving a single 210 MWe steam turbine. The fuel is 9% enriched (520,000 elements) giving 80 GWd/t discharge burn-up. Core outlet temperature is 750ºC. The size was reduced to 250 MWt from earlier 458 MWt modules in order to retain the same core configuration as the prototype HTR-10 and avoid moving to an annular design like South Africa's PBMR (see section on PBMR below). This 210 MWe Shidaowan demonstration plant at Rongcheng in Shandong province is to pave the way for an 18-unit (3x6x210MWe) full-scale power plant on the same site, 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 (CNEC) will have a 32.5% stake and Tsinghua University's INET 20% – it being the main R&D contributor. Projected cost is US$ 430 million (but later units falling to US$1500/kW with generating cost about 5 ¢/kWh). Start-up is scheduled for 2013. The HTR-PM rationale is both eventually to replace conventional reactor technology for power, and also to provide for future hydrogen production. INET is in charge of R&D, and is aiming to increase the size of the 250 MWt module and also utilize thorium in the fuel. Eventually a series of HTRs, possibly with Brayton cycle directly driving the gas turbines, will be factory-built and widely installed throughout China.

Performance of both this and South Africa's PBMR 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 a 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. (See also section on Shidaowan HTR-PM in the information page on Nuclear Power in China and the Research and development section in the information page on China's Nuclear Fuel Cycle.)

PBMR

South Africa's pebble bed modular reactor (PBMR) draws on German expertise and aims to achieve a step change in safety, economics and proliferation resistance. Full-scale production units were planned to be 400 MWt (165 MWe), the prototype being known as the PBMR Demonstration Power Plant (DPP), which was expected to start construction at Koeberg in 2009 and achieve criticality in 2013. Following a series of delays on the DPP project, it was decided to change to a 200 MWt (80 MWe) design[6]. Financial constraints led to further delays[7] and later, in September 2010, the South African government confirmed it would stop funding the project[8].

The earlier plans for the 400 MWt PBMR envisaged a direct cycle (Brayton cycle) gas turbine generator and thermal efficiency about 41%, the helium coolant leaving the bottom of the core at about 900°C and driving a turbine. Power is adjusted by changing the pressure in the system. The helium is passed through a water-cooled pre-cooler and intercooler before being returned to the reactor vessel.

The 200 MWt (80 MWe) later design uses a conventional Rankine cycle, enabling the PBMR to deliver super-heated steam via a steam generator as well as generate electricity. This design "is aimed at steam process heat applications operating at 720°C, which provides the basis for penetrating the nuclear heat market as a viable alternative for carbon-burning, high-emission heat sources."[9] An agreement with Mitsubishi Heavy Industries to take forward the R&D on this design was signed in February 2010. MHI had been involved in the project since 2001, having done the basic design and R&D of the helium-driven turbo generator system and core barrel assembly, the major components of the 400 MWt direct-cycle design.

The PBMR has a vertical steel reactor pressure vessel which contains and supports a metallic core barrel, which in turn supports the cylindrical pebble fuel core. This core is surrounded on the side by an outer graphite reflector and on top and bottom by graphite structures which provide similar upper and lower neutron reflection functions. Vertical borings in the side reflector are provided for the reactivity control elements. Some 360,000 fuel pebbles (silicon carbide-coated 9.6% enriched uranium dioxide particles encased in graphite spheres of 60 mm diameter) cycle through the reactor continuously (about six times each) until they are expended after about three years. This means that a reactor would require 12 total fuel loads in its design lifetime.

For more information on the PBMR, see the PBMR section in the information page on *Nuclear Power in South Africa*.

GT-MHR

A larger US design, the Gas Turbine - Modular Helium Reactor (GT-MHR), will be built as modules of up to 600 MWt. In its electrical application each would directly drive a gas turbine at 47% thermal efficiency, giving 285 MWe capacity. It can also be used for hydrogen production (100,000 t/yr claimed) and other high temperature process heat applications. The annular core consists of 102 hexagonal fuel element columns of graphite blocks with channels for helium coolant and control rods. Graphite reflector blocks are both inside and around the core. Half the core is replaced every 18 months. Burn-up is up to 220 GWd/t, and coolant outlet temperature is 850°C with a target of 1000°C.

The GT-MHR is being developed by General Atomics in partnership with Russia's OKBM Afrikantov, supported by Fuji (Japan). Areva was formerly involved. Initially it was to be used to burn pure ex-weapons plutonium at Seversk (Tomsk) in Russia. A burnable poison such as Er-167 is needed for this fuel. The preliminary design stage was completed in 2001, but the program to construct a prototype in Russia has languished since.

General Atomics says that the GT-MHR neutron spectrum is such, and the TRISO fuel is so stable, that the reactor can be powered fully with separated transuranic wastes (neptunium, plutonium, americium and curium) from light water reactor used fuel. The fertile actinides would enable reactivity control and very high burn-up could be achieved with it – over 500 GWd/t – the 'Deep Burn' concept. Over 95% of the Pu-239 and 60% of other actinides would be destroyed in a single pass.

A smaller version of the GT-MHR, 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.

EM2

In February 2010, General Atomics announced a modified version of its GT-MHR design – the Energy Multiplier Module (EM2). The EM2 is a 500 MWt, 240 MWe helium-cooled fast-neutron HTR operating at 850°C and fuelled with 20 tonnes of used PWR fuel or depleted uranium, plus 22 tonnes of low-enriched uranium (~12% U-235) as starter. Used fuel from this is processed to remove fission products (about 4 tonnes) and the balance is recycled as fuel for subsequent rounds, each time topped up with 4 tonnes of further used PWR fuel. (The means of reprocessing to remove fission products is not specified.) Each refuelling cycle may be as long as 30 years. With repeated recycling the amount of original natural uranium (before use by PWR) used goes up from 0.5% to 50% at about cycle 12. High-level wastes are about 4% of those from PWR on open fuel cycle. A 48% thermal efficiency is claimed, using Brayton cycle. EM2 would also be suitable for process heat applications. The main pressure vessel can be trucked or railed to the site, and installed below ground level.

The company anticipates a 12-year development and licensing period, which is in line with the 80 MWt experimental technology demonstration gas-cooled fast reactor (GFR) in the Generation IV program[l].

Antares

Another full-size HTR design is the Antares reactor being put forward by Areva. It is based on the GT-MHR and has also involved Fuji. Reference design is 600 MWt with prismatic block fuel like the GT-MHR. Target core outlet temperature is 1000°C for a very high temperature reactor (VHTR) version, or up to 850°C for the HTR version. It uses an indirect cycle, possibly with a helium-nitrogen mix in the secondary system, removing the possibility of contaminating the generation or hydrogen production plant with radionuclides from the reactor core.

Adams Engine

A small HTR concept is the Adams Atomic Engines' 10 MWe direct simple Brayton cycle plant with low-pressure nitrogen as the reactor coolant and working fluid, and graphite moderation. The reactor core is a fixed, annular bed with about 80,000 fuel elements each 6 cm diameter and containing approximately 9 grams of heavy metal as TRISO particles, 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%. Power output is controlled by limiting coolant flow. A demonstration plant is proposed for completion after 2018. The Adams Engine is deigned to be competitive with combustion gas turbines.

MTSPNR

A small Russian HTR which was being developed by the N.A. Dollezhal Research and Development Institute of Power Engineering (NIKIET) is the modular transportable small power nuclear reactor (MTSPNR) for heat and electricity supply of remote regions. It is described as a single circuit air-cooled HTR with closed cycle gas turbine. It uses 20% enriched fuel and is designed to run for 25 years without refuelling. A twin unit plant delivers 2 MWe and/or 8 GJ/hr. No recent information is available.

Liquid metal-cooled fast neutron reactors

Fast neutron reactors are designed to use the full energy potential of uranium, rather than about one percent of it that conventional power reactors use. They 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 power regulation is achieved due to the reactivity feedback – loss of coolant flow leads to higher core temperature which slows the reaction. Fast reactors typically use boron carbide control rods.

A Gas-cooled Fast Reactor (GFR) concept – the EM2 – has been announced by General Atomics and is described in the HTR section above.

Hyperion Power Module

The Hyperion Power Module is a 70 MWt/25 MWe lead-bismuth cooled reactor concept using 20% enriched uranium nitride fuel. The reactor was originally conceived as a potassium-cooled self-regulating 'nuclear battery' fuelled by uranium hydride[m]. However, in 2009, Hyperion Power changed the design to uranium nitride fuel and lead-bismuth cooling to expedite design certification[11]. This now classes it as a fast neutron reactor, without moderation. Hyperion claims that the ceramic nitride fuel has superior thermal and neutronic properties compared with uranium oxide. It would be installed below ground level.

The reactor vessel housing the core and primary heat transfer circuit is about 1.5 metres wide and 2.5 metres high. It is easily portable, sealed and has no moving parts. A secondary cooling circuit transfers heat to an external steam generator. The reactor module is designed to operate for electricity or process heat (or cogeneration) continuously for up to 10 years without refuelling. Another reactor module could then take its place in the overall plant. The old module, with fuel burned down to about 15% enrichment, would be put in dry storage at site to cool for up to two years before being returned to the factory.

In March 2010, Hyperion notified the US Nuclear Regulatory Commission that it planned to submit a design certification application in 2012. The company says it has many expressions of interest for ordering units. In September 2010, the company signed an agreement with Savannah River Nuclear Solutions to possibly build a demonstration unit at the Department of Energy site there. (Over 1953-1991, this was where a number of production reactors for weapons plutonium and tritium were built and run.) Hyperion has said it plans to build a prototype by 2015, possibly with uranium oxide fuel if the nitride is not then available.

Encapsulated Nuclear Heat-Source

The Encapsulated Nuclear Heat-Source (ENHS) is a liquid metal-cooled reactor concept of 50 MWe being developed by the University of California, Berkeley. 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 eight 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 automatic. The whole reactor sits in a 17 metre deep silo. Fuel is a uranium-zirconium alloy with 13% 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.

STAR-LM, STAR-H2, SSTAR

The Secure Transportable Autonomous Reactor (STAR) project at Argonne National Laboratory is developing small, multi-purpose systems that operate nearly autonomously for the very long term. The STAR-LM is a factory-fabricated fast neutron modular reactor cooled by lead-bismuth eutectic, with passive safety features. Its 300-400 MWt size means it can be shipped by rail. It uses uranium-transuranic nitride fuel in a 2.5 m diameter cartridge which is replaced every 15 years. Decay heat removal is by external air circulation. The STAR-LM was conceived for power generation with a capacity of about 175 MWe.

The STAR-H2 is an adaptation of the same reactor 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). Its development is further off.

A smaller STAR variant is the Small Sealed Transportable Autonomous Reactor (SSTAR) being developed by Lawrence Livermore, Argonne and Los Alamos National Laboratories in collaboration with others. It has lead or Pb-Bi cooling, 564°C core outlet temperature 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 30-year life without refuelling, the whole reactor unit is then returned for recycling the fuel. The reactor vessel is 12 metre high and 3.2 m diameter (20 MWe version). SSTAR will eventually be coupled to a Brayton cycle turbine using supercritical carbon dioxide. A prototype was envisaged for 2015, but this seems unlikely.

ARC-100

Advanced Reactor Concepts LLC (ARC) is commercializing a 100 MWe sodium-cooled fast reactor based on the 62.5 MWt Experimental Breeder Reactor II (EBR-II). The EBR-II was significant fast reactor prototype at Idaho National Laboratory (formerly Argonne National Laboratory - West) which produced 19 MWe over about 30 years. It used the pyrometallurgically-refined used fuel from light water reactors as fuel, including a wide range of actinides. After operating 1963 to 1994 it is being decommissioned. EBR-II was the basis of the US Integral Fast Reactor (IFR) program (originally the Advanced Liquid Metal Reactor program). An EBR-III of 200-300 MWe was proposed but not developed (see also information page on Fast Neutron Reactors).

The ARC-100 system comprises a uranium alloy core submerged in sodium. The liquid sodium is passed through the core where it is heated to 510°C, then passed through a heat exchanger where it heats sodium in an intermediate loop, which in turn heats working fluid for electricity generation. It would have a refueling interval of 20 years. A 50 MWe version of the ARC is also under development.

LSPR

A lead-bismuth-eutectic (LBE) cooled fast reactor of 150 MWt /53 MWe, the LSPR (LBE-Cooled Long-Life Safe Simple Small Portable Proliferation-Resistant Reactor), is under development in Japan. Fuelled units would be supplied from a factory and operate for 30 years, then be returned. The concept is intended for developing countries.

Rapid-L

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 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 (LEMs) which give burn-up compensation, partial load operation as well as negative reactivity feedback. During normal operation, lithium-6 in the LEM is suspended on an inert gas above the core region. As the reactor temperature rises, the lithium-6 expands, moving the gas/liquid interface down into the core and hence adding negative reactivity. 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.

4S

The Super-Safe, Small & Simple (4S) 'nuclear battery' system is being developed by Toshiba and the Central Research Institute of Electric Power Industry (CRIEPI) in Japan in collaboration with SSTAR work and Westinghouse (owned by Toshiba) in the USA. It uses sodium as coolant (with electromagnetic pumps) and has passive safety features, notably negative temperature coefficient of reactivity. The whole unit would be factory-built, transported to site, installed below ground level, and would drive a steam cycle via a secondary sodium loop. 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 30 MWt (10 MWe) version or 11.5% Pu for the 135 MWt (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 small version; 1.2m diameter and 2.5m high in the larger version) at about one millimetre per week. 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. Burn-up will be 34 GWday/t. 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. A further safety device is a neutron absorber rod which can drop into the core. After 30 years the fuel would be allowed to cool for a year, then it would be removed and shipped for storage or disposal.

Both versions of 4S are designed to automatically maintain an outlet coolant temperature of 550ºC – 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 10 MWe (30 MWt) 4S reactor in that remote location. A pre-application Nuclear Regulatory Commission (NRC) review is under way with a view to application for design certification in October 2010 (delayed from 2009 by NRC workload), and combined construction and operating licence (COL) application to follow. Its design is sufficiently similar to PRISM – GE's modular 150 MWe liquid metal-cooled inherently-safe reactor which went part-way through the NRC approval process (see section below on PRISM) – for it to have good prospects of licensing. Toshiba plans a worldwide marketing program to sell the units for power generation at remote mines, desalination plants and for making hydrogen. Eventually it expects sales for hydrogen production to outnumber those for power supply.

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

PRISM

GE with the US national laboratories had been developing a modular liquid metal-cooled inherently-safe reactor – PRISM (Power Reactor Innovative Small Module) – under the Advanced Liquid Metal Reactor/Integral Fast Reactor (ALMR/IFR) program funded by the US Department of Energy. The program was cancelled in 1994 and no US fast neutron reactor has so far been larger than 66 MWe and none has supplied electricity commercially. However, the 1994 preapplication safety evaluation report[12] for the original PRISM design concluded that "no obvious impediments to licensing the PRISM design had been identified."

Today's PRISM is a GE-Hitachi (GEH) design for compact modular pool-type reactors with passive cooling for decay heat removal. After 30 years of development it represents GEH's Generation IV solution to closing the fuel cycle in the USA. Each PRISM power block consists of two modules of 311 MWe (840 MWt) each, operating at high temperature – over 500°C. The pool-type modules below ground level contain the complete primary system with sodium coolant. The metal Pu & DU fuel is obtained from used light water reactor fuel. However, all transuranic elements are removed together in the electrometallurgical reprocessing so that fresh fuel has minor actinides with the plutonium. Fuel stays in the reactor about six years, with one-third removed every two years, and breeding ratio is 0.8. Used PRISM fuel is recycled after removal of fission products. The commercial-scale plant concept, part of an 'Advanced Recycling Center', would use three power blocks (six reactor modules) to provide 1866 MWe. An application for design certification is expected to be submitted in 2012, and a decision by GEH on building a demonstration plant is expected soon after then. See also Electrometallurgical 'pyroprocessing' section in information page on Processing of Used Nuclear Fuel.

BREST

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 from NIKIET is the BREST fast neutron reactor, of 300 MWe or more with lead as the primary coolant, at 540°C, supplying 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 plutonium can be produced (since there is no uranium blanket), and used fuel can be recycled indefinitely, with on-site facilities. A pilot unit was planned to be built at Beloyarsk, and 1200 MWe units are planned.

SVBR

A smaller and newer Russian design is the Lead-Bismuth Fast Reactor (SVBR) of 75-100 MWe, from Gidropress. This is an integral design, with the steam generators sitting in the same Pb-Bi pool at 400-495°C as the reactor core. It is designed to be able to use a wide variety of fuels, though the reference model uses uranium enriched to 16.5%. Uranium-plutonium fuel is also envisaged. Refuelling interval is 7-8 years. The SVBR-100 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 seven *Alfa*-class submarines, each powered by a compact 155 MWt Pb-Bi cooled reactor, essentially an SVBR, and 70 reactor-years operational experience was acquired with these.)

In December 2009, AKME-Engineering, a 50-50 joint venture, was set up by Rosatom and the En+ Group (a subsidiary of Basic Element Group) to develop and build a pilot SVBR unit[13]. En+ is an associate of EuroSibEnergo and a 53.8% owner of Rusal, which has been in discussion with Rosatom regarding a Far East nuclear power plant and Phase II of the Balakovo nuclear plant. The plan is to complete the design development by 2017 and put on line a 100 MWe pilot facility by 2020, with total investment by Russkiye Mashiny of RUR16 billion ($585 million). The site selection process is underway – earlier plans were to put it Obninsk. The SVBR-100 could be the first reactor cooled by heavy metal to be utilized to generate electricity. It is described by Gidropress as a multi-function reactor.

An SVBR-10 is also envisaged, with the same design principles, a 20-year refuelling interval and generating capacity of 12 MWe, though it too is a multi-purpose unit.

Travelling wave reactor (TWR)

An old design has resurfaced as the travelling wave reactor (TWR). This has been considered in the past as, generically, a candle reactor, or breed-burn reactor, since it burns slowly from one end of a core to the other, making the actual fuel as it goes. The reactor uses natural or depleted uranium packed inside hundreds of hexagonal pillars. In a 'wave' that moves through the core at only one centimetre per year, the U-238 is bred progressively into Pu-239, which is the actual fuel and undergoes fission. The reaction requires a small amount of enriched uranium to get started and could run for decades without refueling. However it is a low-density core and needs to be relatively large. The reactor uses liquid sodium as a coolant, and core temperatures are about 550ºC, giving high thermal efficiency. In 2009 this was selected by *MIT Technology Review* as one of ten emerging technologies of note[14]. In 2010, the company promoting it, Terrapower, made overtures to Toshiba concerning its development, hoping to have a 500 MWe demonstration reactor operating by 2020. Eventual sizes could range from a few hundred MWe to 1000 MWe. Microsoft founder Bill Gates is providing financial backing for Terrapower.

Korean fast reactor designs

In South Korea, the Korea Atomic Energy Research Institute (KAERI) has been working on sodium-cooled fast reactor designs. A second stream of fast reactor development there is via the Nuclear Transmutation Energy Research Centre of Korea (NuTrECK) at Seoul University (SNU). It is working on a lead-bismuth cooled design of 35 MW which would operate on pyro-processed fuel. It is designed to be leased for 20 years and operated without refuelling, then returned to the supplier. It would then be refuelled at the pyro-processing plant and have a design life of 60 years. It would operate at atmospheric pressure, eliminating major concern regarding loss of coolant accidents.

Molten salt reactors

During the 1960s, the USA developed the molten salt breeder reactor concept as the primary back-up option for the fast breeder reactor (cooled by liquid metal) and a small prototype 8 MWt Molten Salt Reactor Experiment (MSRE) operated at Oak Ridge over four years. U-235 fluoride was in molten sodium and zirconium fluorides at 860°C which flowed through a graphite moderator. There is now renewed interest in the concept in Japan, Russia, France and the USA, and one of the six Generation IV designs selected for further development is the molten salt reactor (MSR).

In the MSR, the fuel is a molten mixture of lithium and beryllium fluoride salts with dissolved enriched uranium, thorium or 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[n]. It is not a fast neutron reactor, but with some moderation by the graphite is epithermal (intermediate neutron speed). 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. MSRs have a negative temperature coefficient of reactivity, so will shut down as temperature increases beyond design limits.

Liquid Fluoride Thorium Reactor

The Liquid Fluoride Thorium Reactor (LFTR) is one kind of MSR which breeds its U-233 fuel from a fertile blanket of liquid thorium salts. Some of the neutrons released during fission of the U-233 salt in the reactor core are absorbed by the thorium in the blanket salt. U-233 is thus produced in the blanket and this is then transferred to the fuel salt. LFTRs can rapidly change their power output, and hence be used for load following. Because they are expected to be inexpensive to build and operate, 100 MWe LFTRs could be used as peak and back-up reserve power units.

Fuji MSR

The Fuji MSR is a 100 MWe design to operate 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.

AHTR

The Advanced High-Temperature Reactor (AHTR)[15] is a larger reactor using a coated-particle graphite-matrix fuel like that in the GT-MHR (see above section on the GT-MHR) 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.

Aqueous homogeneous reactors

Aqueous homogeneous reactors (AHRs) have the fuel mixed with the moderator as a liquid. Typically, low-enriched uranium nitrate is in aqueous solution. About 30 AHRs have been built as research reactors and have the advantage of being self-regulating and having the fission products continuously removed from the circulating fuel. However, corrosion problems and the propensity of water to decompose radiolytically (due to fission fragments) releasing gas bubbles have been design problems.

A series of three reactors were built at Los Alamos National Laboratory in the mid-19402/early 1950s. The first AHR at Oak Ridge National Laboratory went critical in 1952, and attained full power of one megawatt in 1953. A second one there reached 5 MW in 1958. Plans for a 70 to 150 MWe commercial unit did not proceed. At Russia's Kurchatov Institute the 20 kW ARGUS AHR has operated since 1981, and R&D on producing Sr-89 and Mo-99 from it is ongoing. The Mo-99 is extremely pure, making the design potentially valuable for its commercial production. As of 2006, only five AHRs were operating, but the concept of extracting medical isotopes directly from the fuel has sparked renewed interest in them. The USA, China and Russia are assessing the prospects of using AHRs for commercial radioisotope production.

In 2008, the IAEA summarised: "The use of solution reactors for the production of medical isotopes is potentially advantageous because of their low cost, small critical mass, inherent passive safety, and simplified fuel handling, processing and purification characteristics. These advantages stem partly from the fluid nature of the fuel and partly from the homogeneous mixture of the fuel and moderator in that an aqueous homogeneous reactor combines the attributes of liquid fuel homogeneous reactors with those of water moderated heterogeneous reactors. If practical methods for handling a radioactive aqueous fuel system are implemented, the inherent simplicity of this type of reactor should result in considerable economic gains in the production of medical isotopes."[16] Thermal power can be 50-300 MW at low temperature and pressure, and low enriched uranium fuel used. However, recovering desired isotopes on a continuous production basis remains to be demonstrated. As well as those in solution, a number of volatile radioisotopes used in nuclear medicine can be recovered from the off-gas arising from radiolytic 'boiling'. For isotopes such a Sr-89 this is very much more efficient than alternative production methods.

Medical Isotope Production System

At the end of 2007, Babcock & Wilcox (B&W) notified the US Nuclear Regulatory Commission that it intended to apply for a licence to construct and operate a Medical Isotope Production System (MIPS) – an AHR system with low-enriched uranium in small 100-200 kW units for Mo-99 production. A single production facility could have four such reactors. B&W expects a five-year lead time to first production. The fuel is brought to criticality in a 200-litre vessel. As fission proceeds, the solution is circulated through an extraction facility to remove the Mo-99 and then back into the reactor vessel, which is at low temperature and pressure. In January 2009, B&W Technical Services Group signed an agreement with radiopharmaceutical and medical device supplier Covidien to develop technology for the MIPS[17].

Modular construction using small reactor units

The IRIS developers have outlined the economic case for modular construction of their design (about 330 MWe), and the argument applies similarly to other smaller units. They point out that IRIS with its size and simple design is ideally suited for modular construction in the sense of progressively building a large power plant with multiple small operating units. The economy of scale is replaced here with the economy of serial production of many small and simple components and prefabricated sections. They expect that construction of the first IRIS unit will be completed in three years, with subsequent reduction to only two years.

Site layouts have been developed with multiple single units or multiple twin units. In each case, units will be constructed so that there is physical separation sufficient to allow construction of the next unit while the previous one is operating and generating revenue. In spite of this separation, the plant footprint can be very compact so that a site with three IRIS single modules providing 1000 MWe capacity is similar or smaller in size than one with a comparable total power single unit.

Eventually IRIS is expected to have a capital cost and production cost comparable with larger plants. But any small unit such as this will potentially have a funding profile and flexibility otherwise impossible with larger plants. As one module is finished and starts producing electricity, it will generate positive cash flow for the next module to be built. Westinghouse estimates that 1000 MWe delivered by three IRIS units built at three year intervals financed at 10% for ten years require a maximum negative cash flow less than $700 million (compared with about three times that for a single 1000 MWe unit). For developed countries small modular units offer the opportunity of building as necessary; for developing countries it may be the only option, because their electric grids cannot take 1000+ MWe single units.

Further Information

Notes

a. Those built as neutron sources are not designed to produce heat or steam, and are less relevant here. [Back]

b. A very general rule is that no single unit should be larger than 15% of grid capacity [Back]

c. Traditional reactor safety systems are 'active' in the sense that they involve electrical or mechanical operation on command. Some engineered systems operate passively, *e.g.* 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. Because small reactors have a higher surface area to volume (and core heat) ratio compared with large units, a lot of the engineering for safety (including heat removal in large reactors) is not needed in the small ones. [Back]

d. In 2010, the American Nuclear Society convened a special committee to look at licensing issues with SMRs in the USA, where dozens of land-based small reactors were built since the 1950s through to the 1980s, proving the safety and security of light water-cooled, gas cooled, and metal cooled SMR technologies. The committee had considerable involvement from SMR proponents, along with the Nuclear Regulatory Commission, Department of Energy laboratories and universities – a total of nearly 50 individuals. The committee's interim report[1] includes the following two tables, which highlight some of the differences between the established US reactor fleet and SMRs.

Comparison of current-generation plant safety systems to potential SMR design

Current generation safety related systems	SMR safety systems
High pressure injection system. Low pressure injection system.	No active safety injection system required. Core cooling is maintained using passive systems.
Emergency sump and associated net positive suction head (NPSH) requirements for safety related pumps.	No safety related pumps for accident mitigation; therefore, no need for sumps and protection of their suction supply.
Emergency diesel generators.	Passive design does not require emergency alternating current (ac) power to maintain core cooling. Core heat removed by heat transfer through vessel.
Active containment heat systems.	None required because of passive heat rejection out of containment.
Containment spray system.	Spray systems are not required to reduce steam pressure or to remove radioiodine from containment.
Emergency core cooling system (ECCS) initiation, instrumentation and control (I&C) systems. Complex systems require significant amount of online testing that contributes to plant unreliability and challenges of safety systems with inadvertent initiations.	Simpler and/or passive safety systems require less testing and are not as prone to inadvertent initiation.
Emergency feedwater system, condensate storage tanks, and associated emergency cooling water supplies.	Ability to remove core heat without an emergency feedwater system is a significant safety enhancement.

Comparison of current-generation plant support systems to potential SMR design

Current LWR support systems	SMR support systems
Reactor coolant pump seals. Leakage of seals has been a safety concern. Seal maintenance and replacement are costly and time consuming.	Integral designs eliminate the need for seals.
Ultimate heat sink and associated interfacing systems. River and seawater systems are active systems, subject to loss of function from such causes as extreme weather conditions and bio fouling.	SMR designs are passive and reject heat by conduction and convection. Heat rejection to an external water heat sink is not required.
Closed cooling water systems are required to support safety related systems for heat removal of core and equipment heat.	No closed cooling water systems are required for safety related systems.
Heating, ventilating, and air conditioning (HVAC). Required to function to support proper operation of safety related systems.	The plant design minimizes or eliminates the need for safety related room cooling eliminating both the HVAC system and associated closed water cooling systems.

[Back]

e. The first two-unit VBER-300 plant is planned to be built in Aktau city, western Kazakhstan, with completion of the first unit envisaged in 2016, and 2017 for the second. The Kazakhstan-Russian Nuclear Stations joint stock company (JSC) was established by Kazatomprom and Atomstroyexport (on a 50:50 basis) in October 2006 for the design, construction and international marketing of the VBER-300. See page on the VBER-300 on the Kazatomprom website (www.kazatomprom.kz) [Back]

f. The 200 MWt (50 MWe net) Melekess VK-50 prototype BWR in Dimitrovgrad, Ulyanovsk commenced operation in 1965 [Back]

g. See the Invap website (www.invap.com.ar) [Back]

h. The page on the NHR-5 on the website of Tsingua University's Institute of Nuclear Energy Technology (now the Institute of Nuclear and New Energy Technology, www.inet.tsinghua.edu.cn) describes the NHR-5 as "a vessel type light water reactor with advanced features, including integral arrangement, natural circulation, hydraulic control rod driving and passive safety systems. Many experiments have been conducted on the NHR-5, such as heat-electricity cogeneration, air-conditioning and seawater desalination." [Back]

i. See the page on Modular Nuclear Reactors on the Babcock & Wilcox website (www.babcock.com) [Back]

j. The 69 fuel assemblies are identical to normal PWR ones, but at about 1.7 m long, a bit less than half the length. [Back]

k. Between 1966 and 1988, the **AVR** experimental pebble bed reactor at Jülich, Germany, operated for over 750 weeks at 15 MWe, most of the time with thorium-based fuel (mixed with high-enriched uranium). The fuel consisted of about 100,000 billiard ball-sized fuel elements. Maximum burn-ups of 150 GWd/t were achieved. It was used to demonstrate the inherent safety of the design due to negative temperature coefficient: reactor power fell rapidly when helium coolant flow was cut off.

The 300 MWe **THTR** (Thorium High-Temperature 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 THTR unsuccessful, though the basic concept was again proven. It drove a steam turbine.

The 200 MWt (72 MWe) **HTR-modul** was then designed by Siemens/Interatom 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 the pebble bed modular reactor (PBMR).

During 1970s and 1980s Nukem manufactured more than 250,000 fuel elements for the AVR and more than one million for the THTR. In 2007, Nukem reported that it had recovered the expertise for this and was making it available as industry support. [Back]

l. The 80 MWt ALLEGRO demonstration GFR is planned by Euratom to incorporate all the architecture and the main materials and components foreseen for the full-sized GFR but without the direct (Brayton) cycle power conversion system. It is being developed mainly by France, with Japan and Switzerland, and operation about 2020 is envisaged. [Back]

m. The Hyperion Power Module was originally designed by Los Alamos National Laboratory as a 70 MWt 'nuclear battery' that uses uranium hydride (UH_3) fuel, which also functions as a moderator. UH_3 stores vast quantities of hydrogen, but this stored hydrogen dissociates as the temperature rises above the operating temperature of 550°C. The release of hydrogen gas lowers the density of the UH_3, which in turn decreases reactivity. This process is reversed as the core temperature drops, leading to the reabsorption of hydrogen. The consequent increase in moderator density results in an increase in core reactivity[10]. All this is without much temperature change since the main energy gain or loss is involved in phase change. [Back]

n. As MSRs will normally operate at much higher temperatures than LWRs, they have potential for process heat. Another option is to have a secondary helium coolant in order to generate power via the Brayton cycle. [Back]

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