World Nuclear Association

Advanced Nuclear Power Reactors

(Updated January 2011)

- **The next two generations of nuclear reactors are currently being developed in several countries.**
- **The first (3rd generation) advanced reactors have been operating in Japan since 1996. Late 3rd generation designs are now being built.**
- **Newer advanced reactors have simpler designs which reduce capital cost. They are more fuel efficient and are inherently safer.**

The nuclear power industry has been developing and improving reactor technology for more than five decades and is starting to build the next generation of nuclear power reactors to fill new orders.

Several generations of reactors are commonly distinguished. Generation I reactors were developed in 1950-60s, and outside the UK none are still running today. Generation II reactors are typified by the present US and French fleets and most in operation elsewhere. Generation III (and 3+) are the Advanced Reactors discussed in this paper. The first are in operation in Japan and others are under construction or ready to be ordered. Generation IV designs are still on the drawing board and will not be operational before 2020 at the earliest.

About 85% of the world's nuclear electricity is generated by reactors derived from designs originally developed for naval use. These and other second-generation nuclear power units have been found to be safe and reliable, but they are being superseded by better designs.

Reactor suppliers in North America, Japan, Europe, Russia and elsewhere have a dozen new nuclear reactor designs at advanced stages of planning, while others are at a research and development stage. Fourth-generation reactors are at concept stage.

Third-generation reactors have:

- a standardised design for each type to expedite licensing, reduce capital cost and reduce construction time,
- a simpler and more rugged design, making them easier to operate and less vulnerable to operational upsets,
- higher availability and longer operating life - typically 60 years,
- further reduced possibility of core melt accidents,*
- resistance to serious damage that would allow radiological release from an aircraft impact,
- higher burn-up to reduce fuel use and the amount of waste,
- burnable absorbers ("poisons") to extend fuel life.

* The US NRC requirement for calculated core damage frequency is $1x10^{-4}$, most current US plants have about $5x10^{-5}$ and Generation III plants are about ten times better than this. The IAEA safety target for future plants is $1x10^{-5}$. Calculated large release frequency (for radioactivity) is generally about ten times less than CDF.

The greatest departure from second-generation designs is that many incorporate passive or inherent safety features* which require no active controls or operational intervention to avoid accidents in the event of malfunction, and may rely on gravity, natural convection or resistance to high temperatures.

* 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. They function without operator control and despite any loss of auxiliary power. 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, but these terms are not properly used to characterise whole reactors.

Another departure is that some PWR types will be designed for load-following. While most French reactors today are operated in that mode to some extent, the EPR design has better capabilities. It will be able to maintain its output at 25% and then ramp up to full output at a rate of 2.5% of rated power per minute up to 60% output and at 5% of rated output per minute up to full rated power. This means that potentially the unit can change its output from 25% to 100% in less than 30 minutes, though this may be at some expense of wear and tear.

Many are larger than predecessors. Increasingly they involve international collaboration.

However, certification of designs is on a national basis, and is safety-based. In Europe there are moves towards harmonised requirements for licensing. In Europe, reactors may also be certified according to compliance with European Utilities Requirements (EUR) of 12 generating companies, which have stringent safety criteria. The EUR are basically a utilities' wish list of some 5000 items needed for new nuclear plants. Plants certified as complying with EUR include Westinghouse AP1000, Gidropress' AES-92, Areva's EPR, GE's ABWR, Areva's Kerena, and Westinghouse BWR 90.

European regulators are increasingly requiring large new reactors to have some kind of core catcher or similar device, so that in a full core-melt accident there is enhanced provision for cooling the bottom of the reactor pressure vessel or simply catching any material that might melt through it. The EPR and VVER-1200 have core-catchers under the pressure vessel, the AP1000 and APWR have provision for enhanced water cooling.

In the USA a number of reactor types have received Design Certification (see below) and others are in process: ESBWR from GE-Hitachi, US EPR from Areva and US-APWR from Mitsubishi. The ESBWR is on track to receive certification about September 2011, and the US EPR in mid 2012. Early in 2008 the NRC said that beyond these three, six pre-application reviews could possibly get underway by about 2010. These included: ACR from Atomic Energy of Canada Ltd (AECL), IRIS from Westinghouse, PBMR from Eskom and 4S from Toshiba as well as General Atomics' GT-MHR apparently. However, for various reasons these seem to be inactive.

Longer term, the NRC expected to focus on the Next-Generation Nuclear Plant (NGNP) for the USA (see US Nuclear Power Policy paper) - essentially the Very High Temperature Reactor (VHTR) among the Generation IV designs.

Joint Initiatives

Two major international initiatives have been launched to define future reactor and fuel cycle technology, mostly looking further ahead than the main subjects of this paper:
Generation IV International Forum (GIF) is a US-led grouping set up in 2001 which has identified six reactor concepts for further investigation with a view to commercial deployment by 2030. See Generation IV paper and DOE web site on "4th generation reactors".

The IAEA's International Project on Innovative Nuclear Reactors and Fuel Cycles (INPRO) is focused more on developing country needs, and initially involved Russia rather than the USA, though the USA has now joined it. It is now funded through the IAEA budget.

At the commercial level, by the end of 2006 three major Western-Japanese alliances had formed to dominate much of the world reactor supply market:

- Areva with Mitsubishi Heavy Industries (MHI) in a major project and subsequently in fuel fabrication,
- General Electric with Hitachi as a close relationship: GE Hitachi Nuclear Energy (GEH)*
- Westinghouse had become a 77% owned subsidiary of Toshiba (with Shaw group 20%).

* GEH is the main international partnership, 60% GE. In Japan it is Hitachi GE, 80% owned by Hitachi.

Subsequently there have been a number of other international collaborative arrangements initiated among reactor vendors and designers, but it remains to be seen which will be most significant.

US Design certification

In the USA, the federal Department of Energy (DOE) and the commercial nuclear industry in the 1990s developed four advanced reactor types. Two of them fall into the category of large "evolutionary" designs which build directly on the experience of operating light water reactors in the USA, Japan and Western Europe. These reactors are in the 1300 megawatt range.

One is an advanced boiling water reactor (**ABWR**) derived from a General Electric design and now promoted both by GE-Hitachi and Toshiba as a proven design, which is in service.

The other type, **System 80+,** is an advanced pressurised water reactor (PWR), which was ready for commercialisation but is not now being promoted for sale. Eight System 80 reactors in South Korea incorporate many design features of the System 80+, which is the basis of the Korean Next Generation Reactor program, specifically the APR-1400 which is expected to be in operation from 2013 and is being marketed worldwide.

The US Nuclear Regulatory Commission (NRC) gave final design certification for both in May 1997, noting that they exceeded NRC "safety goals by several orders of magnitude". The ABWR has also been certified as meeting European utility requirements for advanced reactors. Both GE Hitachi and Toshiba in 2010 submitted separate applications to renew the design certification for their respective versions of ABWR (Toshiba's incorporating design changes submitted to NRC already in connection with application for the South Texas Project). The Japanese version of it differs in allowing modular construction, so is not identical to that licenced in the USA.

Another, more innovative US advanced reactor is smaller - 600 MWe - and has passive safety features (its projected core damage frequency is more than 100 times less than today's NRC requirements). The Westinghouse **AP600** gained NRC final design certification in 1999 (AP = Advanced Passive).

These NRC approvals were the first such generic certifications to be issued and are valid for 15 years. As a result of an exhaustive public process, safety issues within the scope of the certified designs have been fully resolved and hence will not be open to legal challenge during licensing for particular plants. US utilities will be able to obtain a single NRC licence to both construct and operate a reactor before construction begins.

Separate from the NRC process and beyond its immediate requirements, the US nuclear industry selected one standardised design in each category - the large ABWR and the medium-sized AP600, for detailed first-of-a-kind engineering (FOAKE) work. The US$ 200 million program was half funded by DOE and means that prospective buyers now have fuller information on construction costs and schedules.

The 1100 MWe-class Westinghouse **AP1000**, scaled-up from the AP600, received final design certification from the NRC in December 2005 - the first Generation 3+ type to do so. It represented the culmination of a 1300 man-year and $440 million design and testing program. In May 2007 Westinghouse applied for UK generic design assessment (pre-licensing approval) based on the NRC design certification, and expressing its policy of global standardisation. The application was supported by European utilities.

Overnight capital costs were originally projected at $1200 per kilowatt and modular design is expected to reduce construction time eventually to 36 months. The AP1000 generating costs are also expected to be very competitive and it has a 60-year operating life. It is being built in China (4 units under construction, with many more to follow) and is under active consideration for building in Europe and USA. It is capable of running on a full MOX core if required.

In February 2008 the NRC accepted an application from Westinghouse to amend the AP1000 design, and this review is expected to be complete in September 2011.

A contrast between the 1188 MWe Westinghouse reactor at Sizewell B in the UK and the Generation III+ AP1000 of similar-power illustrates the evolution from Generation II types. First, the AP1000 footprint is very much smaller - about one quarter the size, secondly the concrete and steel requirements are less by a factor of five*, and thirdly it has modular construction. A single unit will have 149 structural modules of five kinds, and 198 mechanical modules of four kinds: equipment, piping & valve, commodity, and standard service modules. These comprise one third of all construction and can be built off site in parallel with the on-site construction.

*Sizewell B: 520,000 m3 concrete (438 m3/MWe), 65,000 t rebar (55 t/MWe);
AP1000: <1000,000 m3 concrete (90 m3/MWe, <12,000 t rebar (11 t/MWe).

At Sanmen in China, where the first AP1000 units are under construction, the first module - of 840 tonnes - has been lifted into place. More than 50 other modules to be used in the reactors' construction weigh more than 100 tonnes, while 18 weigh in excess of 500 tonnes.

Light Water Reactors

EPR

Areva NP (formerly Framatome ANP) has developed a large (4590 MWt, typically 1750 MWe gross and 1630 MWe net) European pressurised water reactor (**EPR**), which was confirmed in mid 1995 as the new standard design for France and received French design approval in 2004. It is a 4-loop design derived from the German Konvoi types with features from the French N4, and is expected to provide power about 10% cheaper than the N4. It has several active safety systems, and a core catcher under the pressure vessel. It will operate flexibly to follow loads, have fuel burn-up of 65 GWd/t and a high thermal efficiency, of 37%, and net efficiency of 36%. It is capable of using a full core load of MOX. Availability is expected to be 92% over a 60-year service life. It has four separate, redundant safety systems rather than passive safety.

The first EPR unit is being built at Olkiluoto in Finland, the second at Flamanville in France, the third European one will be at Penly in France, and two further units are under construction at Taishan in China.

A US version, the **US-EPR** quoted as 1710 MWe gross and about 1580 MWe net, was submitted for US design certification in December 2007, and this is expected to be granted early 2012. The first unit (with 80% US content) is expected to be grid connected by 2020. It is now known as the Evolutionary PWR (EPR). Much of the one million man-hours of work involved in developing this US EPR is making the necessary changes to output electricity at 60 Hz instead of the original design's 50 Hz. The main development of the type is to be through UniStar Nuclear Energy, but other US proposals also involve it.

AP1000

The Westinghouse AP1000 is a 2-loop PWR which has evolved from the smaller AP600, one of the first Generation III reactor designs certified by the US NRC, in 2005. Simplification was a major design objective of the AP1000, in overall safety systems, normal operating systems, the control room, construction techniques, and instrumentation and control systems provide cost savings with improved safety margins. Core damage frequency is 5x10-7. It has a passive core cooling system including passive residual heat removal, improved containment isolation, passive containment cooling system and in-vessel retention of core damage. It is being built in China, and the Vogtle site is being prepared for initial units in USA. The first four units are on schedule, being assembled from modules. It is quoted as 1200 MWe gross and 1117 MWe net (3400 MWt), though 1250 MWe gross in China. Westinghouse earlier claimed a 36 month construction time to fuel loading, but the first ones being built in China are on a 51 month timeline to fuel loading, or 57 month schedule to grid connection.

ABWR

The advanced boiling water reactor (ABWR) is derived from a General Electric design. Two examples built by Hitachi and two by Toshiba are in commercial operation in Japan (1315 MWe net), with another two under construction there and two in Taiwan. Four more are planned in Japan and another two in the USA. It is basically a 1380 MWe (gross) unit (3926 MWt in Toshiba version), though GE Hitachi quote 1350-1600 MWe net and Hitachi is also developing 600, 900 and 1700 MWe versions of it. Toshiba outlines development from 1350 MWe class of a 1600-1700 MWe class as well as an 800-1000 MWe class derivatives. Tepco is funding the design of a next generation BWR, and the ABWR-II is quoted as 1717 MWe.

The first four ABWRs were each built in 39 months on a single-shift basis. Though GE and Hitachi have subsequently joined up, Toshiba retains some rights over the design, as does Tepco. Both GE-Hitachi and Toshiba (with NRG Energy in USA) are marketing the design. Design life is 60 years.

ESBWR

GE Hitachi Nuclear Energy's **ESBWR** is a Generation III+ technology that utilizes passive safety features and natural circulation principles and is essentially an evolution from a predecessor design, the SBWR at 670 MWe. GE says it is safer and more efficient than earlier models, with 25% fewer pumps, valves and motors. The ESBWR (4500 MWt) will produce approximately 1600 MWe gross, and 1535 MWe net, depending on site conditions, and has a design life of 60 years. It was more fully known as the Economic & Simplified BWR (ESBWR) and leverages proven technologies from the ABWR. The ESBWR is in advanced stages of licensing review with the US NRC for GE Hitachi and is on schedule for full design certification in 2010-11. Core damage frequency is quoted as $1x10^{-8}$.

GEH is selling this alongside the ABWR, which it characterises as more expensive to build and operate, but proven. ESBWR is more innovative, with lower building and operating costs and a 60-year life.

APWR

Mitsubishi's large APWR - advanced PWR of 1538 MWe gross (4451 MWt) - was developed in collaboration with four utilities (Westinghouse was earlier involved). The first two are planned for Tsuruga, coming on line from 2016. It is a 4-loop design with 257 fuel assemblies and neutron reflector, is simpler, combines active and passive cooling systems to greater effect, and has over 55 GWd/t (and up to 62 GWd/t) fuel burn-up. It is the basis for the next generation of Japanese PWRs. The planned APWR+ is 1750 MWe and has full-core MOX capability.

The **US-APWR** will be 1700 MWe gross, about 1620 MWe net, due to longer (4.3m) fuel assemblies, higher thermal efficiency (37%) and has 24 month refuelling cycle. US design certification application was in January 2008 with approval expected in 2011 and certification mid 2012. In March 2008 MHI submitted the same design for EUR certification, as **EU-APWR,** and it will join with Iberdrola Engineering & Construction in bidding for sales of this in Europe. Iberdrola would be responsible for building the plants.

The Japanese government is expected to provide financial support fort US licensing of both US-APWR and the ESBWR. The Washington Group International will be involved in US developments with Mitsubishi Heavy Industries (MHI). The US-APWR has been selected by Luminant for Comanche Peak, Texas, and when the COL application for the new reactors was lodged Luminant and MHI announced a joint venture to build and own the twin-unit plant. This Comanche Peak Nuclear Power Co is 88% Luminant, 12% MHI.

APR1400

South Korea's APR-1400 Advanced PWR design has evolved from the US System 80+ with enhanced safety and seismic robustness and was earlier known as the Korean Next-Generation Reactor. Design certification by the Korean Institute of Nuclear Safety was awarded in May 2003. It is 1455 MWe gross, 1350-1400 MWe net (3983 MWt) with 2-loop primary circuit. The first of these is under construction - Shin-Kori-3 & 4, expected to be operating in 2013. Fuel has burnable poison and will have up to 55 GWd/t burn-up, refueling cycle c 18 months, outlet temperature 324ºC. Projected cost at the end of 2009 was US$ 2300 per kilowatt, with 48-month construction time. Plant life is 60 years, seismic design basis is 300 Gal. A low-speed (1800 rpm) turbine is envisaged. It has been chosen as the basis of the United Arab Emirates nuclear program on the basis of cost and reliable building schedule, and an application for US Design Certification is planned in 2012.

Based on this there are plans for an EU version (EU-APR1400) and a more advanced 1550 MWe (gross) Generation III+ version, the APR+. In addition some of the APR features are being incorporated into a development of the OPR-1000 to give an exportable APR-1000.

Atmea1

The Atmea 1 is developed by the Atmea joint venture established in 2006 by Areva NP and Mitsubishi Heavy Industries to produce an evolutionary 1150 MWe net 93150 MWt) three-loop PWR using the same steam generators as EPR. This has extended fuel cycles, 37% net thermal efficiency, 157 fuel assemblies 4.2 m long, 60-year life, and the capacity to use mixed-oxide fuel only. Fuel cycle is flexible 12 to 24 months with short refuelling outage and the reactor has load-following and frequency control capability. The partners are submitting this to French regulator ASN for safety review, which is expected to be complete in late 2011. The reactor is regarded as mid-sized relative to other generation III units and will be marketed primarily to countries embarking upon nuclear power programs.

Kerena

Together with German utilities and safety authorities, Areva NP is also developing another evolutionary design, the Kerena, a 1290 MWe gross, 1250 MWe net (3370 MWt) BWR with 60-year design life formerly known as **SWR 1000**,. The design, based on the Gundremmingen plant built by Siemens, was completed in 1999 and US certification was sought, but then deferred. As well as many passive safety features,including a core-catcher, the reactor is simpler overall and uses high-burnup fuels enriched to 3.54%, giving it refuelling intervals of up to 24 months. It has 37% net efficiency and is ready for commercial deployment.

AES-92, V392

Gidropress late-model VVER-1000 units with enhanced safety (AES 92 & 91 power plants) are being built in India and China. Two more are planned for Belene in Bulgaria. The **AES-92** is certified as meeting EUR, and its V-392 reactor is considered Generation III. They have four coolant loops and are rated 3000 MWt.

AES-2006, MIR-1200

A third-generation standardised **VVER-1200** (V-491) reactor of 1170 MWe net, possibly 1290 MWe gross and 3200 MWt is in the AES-2006 plant. It is an evolutionary development of the well-proven VVER-1000 in the AES-92 plant, with longer life (50, not nominal 30 years), greater power, and greater efficiency (36.56% instead of 31.6%) and up to 70 GWd/t burn-up. They retain four coolant loops. The lead units are being built at Novovoronezh II, to start operation in 2012-13 followed by Leningrad II for 2013-14. An AES-2006 plant will consist of two of these OKB Gidropress reactor units expected to run for 50 years with capacity factor of 90%. Ovrnight capital cost was said to be US$ 1200/kW and construction time 54 months. They have enhanced safety including that related to earthquakes and aircraft impact with some passive safety features, double containment and core damage frequency of $1x10^{-7}$.

Atomenergoproekt say that the AES-2006 conforms to both Russian standards and European Utilities Requirements (EUR). In Europe the basic technology is being called the Europe-tailored reactor design, **MIR-1200** (Modernised International Reactor) with some Czech involvement.

The VVER-1500 model was being developed by Gidropress. It will have 45-55 and up to 60 MWd/t burn-up and enhanced safety, giving 1500 MWe gross from 4250 MWt. Design was expected to be complete in 2007 but the project was shelved in favour of the evolutionary VVER-1200.

IRIS

Another US-origin but international project which is a few years behind the AP1000 is the IRIS (International Reactor Innovative & Secure). Westinghouse is leading a wide consortium developing it as an advanced 3rd Generation project. IRIS is a modular 335 MWe pressurised water reactor with integral steam generators and primary coolant system all within the pressure vessel. It is nominally 335 MWe but can be less, eg 100 MWe. Fuel is initially similar to present LWRs with 5% enrichment and burnable poison, in fact fuel assemblies are "identical to those ... in the AP1000". These would have burn-up of 60 GWd/t with fuelling interval of 3 to 3.5 years, but IRIS is designed ultimately for fuel with 10% enrichment and 80 GWd/t burn-up with an 8-year cycle, or equivalent MOX core. The core has low power density. IRIS could be deployed in the next decade, and US design certification is at pre-application stage. Estonia has expressed interest in building a pair of them. Multiple modules are expected to cost US$ 1000-1200 per kW for power generation, though some consortium partners are interested in desalination, one in district heating.

VBER-300

OKBM's **VBER-300** PWR is a 295-325 MWe unit (917 MWt) developed from naval power plants and was originally envisaged in pairs as a floating nuclear power plant. It is designed for 60 year life and 90% capacity factor. It now planned to develop it as a land-based unit with Kazatomprom, with a view to exports, and the first unit will be built in Kazakhstan.

The VBER-300 and the similar-sized VK300 are more fully described in the Small Nuclear Power Reactors paper.

RMWR

The Reduced-Moderation Water Reactor (RMWR) is a light water reactor, essentially as used today, with the fuel packed in more tightly to reduce the moderating effect of the water. Considering the BWR variant (resource-renewable BWR - RBWR), only the fuel assemblies and control rods are different. In particular, the fuel assemblies are much shorter, so that they can still be cooled adequately. Ideally they are hexagonal, with Y-shaped control rods. The reduced moderation means that more fissile plutonium is produced and the breeding ratio is around 1 (instead of about 0.6), and much more of the U-238 is converted to Pu-239 and then burned than in a conventional reactor. Burn-up is about 45 GWd/t, with a long cycle. Initial seed (and possibly all) MOX fuel needs to have about 10% Pu. The void reactivity is negative, as in conventional LWR. A Hitachi RBWR design based on the ABWR-II has the central part of each fuel assembly (about 80% of it) with MOX fuel rods and the periphery uranium oxide. In the MOX part, minor actinides are burned as well as recycled plutonium.

The main rationale for RMWRs is extending the world's uranium resource and providing a bridge to widespread use of fast neutron reactors. Recycled plutonium should be used preferentially in RMWRs rather than as MOX in conventional LWRs, and multiple recycling of plutonium is possible. Japan Atomic Energy Research Institute (JAERI) started the research on RMWRs in 1997 and then collaborated in the conceptual design study with the Japan Atomic Power Company (JAPCO) in 1998. Hitachi have also been closely involved.

A new reprocessing technology is part of the RMWR concept. This is the fluoride volatility process, developed in 1980s, and is coupled with solvent extraction for plutonium to give the Fluorex process. In this, 90-92% of the uranium in the used fuel is volatalised as UF_6, then purified for enrichment or storage. The residual is put through a Purex circuit which separates fission products and minor actinides as high-level waste, leaving the unseparated U-Pu mix (about 4:1) to be made into MOX fuel.

Heavy Water Reactors

In Canada, the government-owned Atomic Energy of Canada Ltd (AECL) has had two designs under development which are based on its reliable CANDU-6 reactors, the most recent of which are operating in China.

The CANDU-9 (925-1300 MWe) was developed from this also as a single-unit plant. It has flexible fuel requirements ranging from natural uranium through slightly-enriched uranium, recovered uranium from reprocessing spent PWR fuel, mixed oxide (U & Pu) fuel, direct use of spent PWR fuel, to thorium. It may be able to burn military plutonium or actinides separated from reprocessed PWR/BWR waste. A two year licensing review of the CANDU-9 design was successfully completed early in 1997, but the design has been shelved.

EC6

Some of the innovation of this, along with experience in building recent Korean and Chinese units, was then put back into the Enhanced CANDU-6 (EC6) - built as twin units - with power increase to 750 MWe gross (690 MWe net, 2084 MWt) and flexible fuel options, plus 4.5 year construction and 60-year plant life (with mid-life pressure tube replacement). This is under consideration for new build in Ontario. AECL claims it as a Generation III design.

The Advanced Candu Reactor (ACR), a 3rd generation reactor, is a more innovative concept. While retaining the low-pressure heavy water moderator, it incorporates some features of the pressurised water reactor. Adopting light water cooling and a more compact core reduces capital cost, and because the reactor is run at higher temperature and coolant pressure, it has higher thermal efficiency.

ACR

The ACR-700 design was 700 MWe but is physically much smaller, simpler and more efficient as well as 40% cheaper than the CANDU-6. But the ACR-1000 of 1080-1200 MWe (3200 MWt) is now the focus of attention by AECL. It has more fuel channels (each of which can be regarded as a module of about 2.5 MWe). The ACR will run on low-enriched uranium (about 1.5-2.0% U-235) with high burn-up, extending the fuel life by about three times and reducing high-level waste volumes accordingly. It will also efficiently burn MOX fuel, thorium and actinides.

Regulatory confidence in safety is enhanced by a small negative void reactivity for the first time in CANDU, and utilising other passive safety features as well as two independent and fast shutdown systems. Units will be assembled from prefabricated modules, cutting construction time to 3.5 years. ACR units can be built singly but are optimal in pairs. They will have 60 year design life overall but require mid-life pressure tube replacement.

ACR is moving towards design certification in Canada, with a view to following in China, USA and UK. In 2007 AECL applied for UK generic design assessment (pre-licensing approval) but then withdrew after the first stage. In the USA, the ACR-700 is listed by NRC as being at pre application review stage. The first ACR-1000 unit could be operating in 2016 in Ontario.

The **CANDU X** or SCWR is a variant of the ACR, but with supercritical light water coolant (eg 25 MPa and 625ºC) to provide 40% thermal efficiency. The size range envisaged is 350 to 1150 MWe, depending on the number of fuel channels used. Commercialisation envisaged after 2020.

AHWR

India is developing the Advanced Heavy Water reactor (AHWR) as the third stage in its plan to utilise thorium to fuel its overall nuclear power program. The AHWR is a 300 MWe gross (284 MWe net, 920 MWt) reactor moderated by heavy water at low pressure. The calandria has about 450 vertical pressure tubes and the coolant is boiling light water circulated by convection. A large heat sink - "Gravity-driven water pool" - with 7000 cubic metres of water is near the top of the reactor building. Each fuel assembly has 30 Th-U-233 oxide pins and 24 Pu-Th oxide pins around a central rod with burnable absorber. Burn-up of 24 GWd/t is envisaged. It is designed to be self-sustaining in relation to U-233 bred from Th-232 and have a low Pu inventory and consumption, with slightly negative void coefficient of reactivity. It is designed for 100-year plant life and is expected to utilise 65% of the energy of the fuel, with two thirds of that energy coming from thorium via U-233.

Once it is fully operational, each AHWR fuel assembly will have the fuel pins arranged in three concentric rings arranged:

Inner: 12 pins Th-U-233 with 3.0% U-233,
Intermediate: 18 pins Th-U-233 with 3.75% U-233,
Outer: 24 pins Th-Pu-239 with 3.25% Pu.

The fissile plutonium content will decrease from an initial 75% to 25% at equilibrium discharge burn-up level.

As well as U-233, some U-232 is formed, and the highly gamma-active daughter products of this confer a substantial proliferation resistance.

In 2009 an export version of this design was announced: the **AHWR-LEU**. This will use low-enriched uranium plus thorium as a fuel, dispensing with the plutonium input. About 39% of the power will come from thorium (via in situ conversion to U-233), and burn-up will be 64 GWd/t. Uranium enrichment level will be 19.75%, giving 4.21% average fissile content of the U-Th fuel. While designed for closed fuel cycle, this is not required. Plutonium production will be less than in light water reactors, and the fissile proportion will be less and the Pu-238 portion three times as high, giving inherent proliferation resistance. The AEC says that "the reactor is manageable with modest industrial infrastructure within the reach of developing countries."

In the AHWR-LEU, the fuel assemblies will be configured:
Inner ring: 12 pins Th-U with 3.555% U-235,
Intermediate ring: 18 pins Th-U with 4.345% U-235,
Outer ring: 24 pins Th-U with 4.444% U-235.

High-Temperature Gas-Cooled Reactors

These reactors use helium as a coolant at up to 950ºC, which either makes steam conventionally or directly drives a gas turbine for electricity and a compressor to return the gas to the reactor core. Fuel is in the form of TRISO particles less than a millimetre in diameter. Each has a kernel of uranium oxycarbide, with the uranium enriched up to 17% 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. These particles may be arranged: in blocks as hexagonal 'prisms' of graphite, or in billiard ball-sized pebbles of graphite encased in silicon carbide.

HTR-PM

The first commercial version will be China's HTR-PM, being built at Shidaowan in Shandong province. It has been developed by Tsinghua University's INET, which is the R&D leader and Chinergy Co., with China Huaneng Group leading the demonstration plant project. This will have two reactor modules, each of 250 MWt/ 105 MWe, using 9% enriched fuel (520,000 elements) giving 80 GWd/t discharge burnup. With an outlet temperature of 750ºC the pair will drive a single steam cycle turbine at about 40% thermal efficiency. This 210 MWe Shidaowan demonstration plant 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.

PBMR

South Africa's **Pebble Bed Modular Reactor** (PBMR) was being developed by a consortium led by the utility Eskom, with Mitsubishi Heavy Industries from 2010. It draws on German expertise. It aims for a step change in safety, economics and proliferation resistance. Production units would be 165 MWe. The PBMR will ultimately have 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. (In the Demonstration Plant it will transfer heat in a steam generator rather than driving a turbine directly.)

Up to 450,000 fuel pebbles recycle through the reactor continuously (about six times each) until they are expended, giving an average enrichment in the fuel load of 4-5% 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 are replaced, giving high capacity factor. Each unit will finally discharge about 19 tonnes/yr of spent pebbles to ventilated on-site storage bins. A reactor will use about 13 fuel loads in a 40-year lifetime. Operational cycles are expected to be six years between shutdowns.

Performance includes great flexibility in loads (40-100%), 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. Overnight capital cost (when in clusters of eight units) is expected to be modest and generating cost very competitive. However, development has ceased due to lack of funds and customers.

GT-MHR

A larger US design, the **Gas Turbine - Modular Helium Reactor** (GT-MHR), is planned 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 100,000 MWd/t. It is being developed by General Atomics in partnership with Russia's OKBM Afrikantov, supported by Fuji (Japan). Initially it was to be used to burn pure ex-weapons plutonium at Seversk (Tomsk) in Russia. The preliminary design stage was completed in 2001, but the program has stalled since.

Areva's Antares is based on the GT-MHR.

Fuller descriptions of HTRs is in the Small Nuclear Power Reactors paper .

Fast Neutron Reactors

Several countries have research and development programs for improved Fast Breeder Reactors (FBR), which are Fast Neutron Reactors (FNR) configured with a conversion or breeding ratio of more than 1 (ie more fissile nuclei are produced than are fissioned). These use the uranium-238 in reactor fuel as well as the fissile U-235 isotope used in most reactors.

About 20 liquid metal-cooled FBRs have already been operating, some since the 1950s, and some have supplied electricity commercially. About 300 reactor-years of operating experience have been accumulated. See also Fast Neutron Reactors paper.

Natural uranium contains about 0.7 % U-235 and 99.3 % U-238. In any reactor the U-238 component is turned into several isotopes of plutonium during its operation. Two of these, Pu 239 and Pu 241, then undergo fission in the same way as U 235 to produce heat. In a fast neutron reactor this process is optimised so that it can 'breed' fuel, often using a depleted uranium blanket around the core. FBRs can utilise uranium at least 60 times more efficiently than a normal reactor. They are however expensive to build and could only be justified economically if uranium prices were to rise to pre-1980 values, well above the current market price.

For this reason research work almost ceased for some years, and that on the 1450 MWe European FBR has apparently lapsed. Closure of the 1250 MWe French Superphenix FBR after very little operation over 13 years also set back developments.

Research continues in India. At the Indira Gandhi Centre for Atomic Research a 40 MWt fast breeder test reactor has been operating since 1985. In addition, the tiny Kamini there is employed to explore the use of thorium as nuclear fuel, by breeding fissile U-233. In 2004 construction of a 500 MWe prototype fast breeder reactor started at Kalpakkam. The unit is expected to be operating in 2012, fuelled with uranium-plutonium oxide (the reactor-grade Pu being from its existing PHWRs) and with a thorium blanket to breed fissile U-233. This will take India's ambitious thorium program to stage 2, and set the scene for eventual full utilisation of the country's abundant thorium to fuel reactors.

Japan plans to develop FBRs, and its Joyo experimental reactor which has been operating since 1977 is now being boosted to 140 MWt. The 280 MWe Monju prototype commercial FBR was connected to the grid in 1995, but was then shut down for 15 years due to a sodium leak. It restarted in 2010.

Mitsubishi Heavy Industries (MHI) is involved with a consortium to build the **Japan Standard Fast Reactor** (JSFR) concept, though with breeding ratio less than 1:1. This is a large unit which will burn actinides with uranium and plutonium in oxide fuel. It could be of any size from 500 to 1500 MWe. In this connection MHI has also set up Mitsubishi FBR Systems (MFBR). The demonstration FR model is due to be committed in 2015 and on line in 2025, and a 1500 MWe commercial FR is proposed by MHI for 2050.

The Russian BN-600 fast breeder reactor at Beloyarsk has been supplying electricity to the grid since 1981 and has the best operating and production record of all Russia's nuclear power units. It uses uranium oxide fuel and the sodium coolant delivers 550°C at little more than atmospheric pressure. The BN 350 FBR operated in Kazakhstan for 27 years and about half of its output was used for water desalination. Russia plans to reconfigure the BN-600 to burn the plutonium from its military stockpiles.

Advanced FNRs include the following:

BN-800

The first BN-800, a new more powerful (880 MWe gross, 2100 MWt) FBR from OKBM with improved features is being built at Beloyarsk. It has considerable fuel flexibility - U+Pu nitride, MOX, or metal, and with breeding ratio up to 1.3. It has much enhanced safety and improved economy - operating cost is expected to be only 15% more than VVER. It is capable of burning 2 tonnes of plutonium per year from dismantled weapons and will test the recycling of minor actinides in the fuel. The BN-800 has been sold to China, and two units are due to start construction there in 2012.

However, the BN-800 is likely to be the last such reactor design built (outside India’s thorium program), with a fertile blanket of depleted uranium around the core. Further fast reactors will have an integrated core to minimise the potential for weapons proliferation from bred Pu-239. Beloyarsk-5 is designated as a BREST design.

BREST

Russia has experimented with several lead-cooled reactor designs, and has used lead-bismuth cooling for 40 years in reactors for its 7 Alfa class submarines. Pb-208 (54% of naturally-occurring lead) is transparent to neutrons. A significant new Russian design from NIKIET is the BREST fast neutron reactor, of 300 MWe or more with lead as the primary coolant, at 540□C, and supercritical steam generators. It is inherently safe and uses a high-density U+Pu nitride fuel with no requirement for high enrichment levels. No weapons-grade plutonium can be produced (since there is no uranium blanket - all the breeding occurs in the core). Also it is an equilibrium core, so there are no spare neutrons to irradiate targets. The initial cores can comprise Pu and spent fuel - hence loaded with fission products, and radiologically 'hot'. Subsequently, any surplus plutonium, which is not in pure form, can be used as the cores of new reactors. Used fuel can be recycled indefinitely, with on-site reprocessing and associated facilities. A pilot unit is planned for Beloyarsk by 2020, and 1200 MWe units are proposed.

ELSY

The European Lead-cooled SYstem (ELSY) of 600 MWe in Europe, led by Ansaldo Nucleare from Italy and financed by Euratom. ELSY is a flexible fast neutron reactor which can use depleted uranium or thorium fuel matrices, and burn actinides from LWR fuel. Liquid metal (Pb or Pb-Bi eutectic) cooling is at low pressure .The design was nearly complete in 2008 and a small-scale demonstration facility is planned. It runs on MOX fuel at 480°C and the molten lead is pumped to eight steam generators, though decay heat removal is passive, by convection.

PRISM

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

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

KALIMER

Korea's KALIMER (Korea Advanced LIquid MEtal Reactor) is a 600 MWe pool type sodium-cooled fast reactor designed to operate at over 500ºC. It has evolved from a 150 MWe version. It has a transmuter core, and no breeding blanket is involved. Future development of KALIMER as a Generation IV type is envisaged.

See also paper on Fast Neutron Reactors.

Generation IV Designs

See paper on six Generation IV Reactors, also DOE paper.

Small Reactors

See also paper on Small Nuclear Power Reactors for other advanced designs, mostly under 300 MWe.

Accelerator-Driven Systems

A recent development has been the merging of accelerator and fission reactor technologies to generate electricity and transmute long-lived radioactive wastes.
A high-energy proton beam hitting a heavy metal target produces neutrons by spallation. The neutrons cause fission in the fuel, but unlike a conventional reactor, the fuel is sub-critical, and fission ceases when the accelerator is turned off. The fuel may be uranium, plutonium or thorium, possibly mixed with long-lived wastes from conventional reactors.

Many technical and engineering questions remain to be explored before the potential of this concept can be demonstrated. *See also* ADS briefing paper.

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The Commission believes designers should consider several reactor characteristics, including:

- Highly reliable, less complex safe shutdown systems, particularly ones with inherent or passive safety features;
- Simplified safety systems that allow more straightforward engineering analysis, operate with fewer operator actions and increase operator comprehension of reactor conditions;
- Concurrent resolution of safety and security requirements, resulting in an overall security system that requires fewer human actions;
- Features that prevent a simultaneous breach of containment and loss of core cooling from an aircraft impact, or that inherently delay any radiological release, and;
- Features that maintain spent fuel pool integrity following an aircraft impact.

Advanced Thermal Reactors being marketed

Country and developer	Reactor	Size MWe gross	Design Progress	Main Features (improved safety in all)
US-Japan (GE-Hitachi, Toshiba)	ABWR	1380	Commercial operation in Japan since 1996-7. In US: NRC certified 1997, FOAKE.	Evolutionary design. More efficient, less waste. Simplified construction (48 months) and operation.
USA (Westinghouse)	AP600 AP1000 (PWR)	600 1200	AP600: NRC certified 1999, FOAKE. AP1000 NRC certification 2005, under construction in China, many more planned there. Amended US NRC certification expected Sept 2011.	Simplified construction and operation. 3 years to build. 60-year plant life.

Europe (Areva NP)	EPR US-EPR (PWR)	1750	Future French standard. French design approval. Being built in Finland, France & China. Undergoing certification in USA.	Evolutionary design. High fuel efficiency. Flexible operation
USA (GE- Hitachi)	ESBWR	1600	Developed from ABWR, undergoing certification in USA, likely constructiion there.	Evolutionary design. Short construction time.
Japan (utilities, Mitsubishi)	APWR US-APWR EU-APWR	1530 1700 1700	Basic design in progress, planned for Tsuruga US design certification application 2008.	Hybrid safety features. Simplified Construction and operation.
South Korea (KHNP, derived from Westinghouse)	APR-1400 (PWR)	1450	Design certification 2003, First units expected to be operating c 2013. Sold to UAE.	Evolutionary design. Increased reliability. Simplified construction and operation.
Europe (Areva NP)	Kerena (BWR)	1250	Under development, pre-certification in USA	Innovative design. High fuel efficiency.
Russia (Gidropress)	VVER-1200 (PWR)	1290	Under construction at Leningrad and Novovoronezh plants	Evolutionary design. High fuel efficiency. 50-year plant life
Canada (AECL)	Enhanced CANDU-6	750	Improved model Licensing approval 1997	Evolutionary design. Flexible fuel requirements.
Canada (AECL)	ACR	700 1080	undergoing certification in Canada	Evolutionary design. Light water cooling. Low-enriched fuel.
China (INET, Chinergy)	HTR-PM	2x105 (module)	Demonstration plant due to start building at Shidaowan	Modular plant, low cost. High temperature. High fuel efficiency.