WORLD NUCLEAR ASSOCIATION

Advanced Nuclear Power Reactors

(Updated February 2020)

- **Improved designs of nuclear power reactors are constantly being developed internationally.**
- **The first so-called Generation III advanced reactors have been operating in Japan since 1996. These have now evolved further.**
- **Newer advanced reactors now being built have simpler designs which are intended to reduce capital cost. They are more fuel efficient and are inherently safer.**
- **Many new designs are small – up to 300 MWe. These are described in a separate information paper.***

** For smaller advanced reactors see the companion paper on Small Nuclear Power Reactors.*

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 the last one shut down in the UK in 2015. Generation II reactors are typified by the present US and French fleets and most in operation elsewhere. So-called Generation III (and III+) are the advanced reactors discussed in this paper, though the distinction from Generation II is arbitrary. The first ones are in operation in Japan and others are under construction in several countries. Generation IV designs are still on the drawing board and will not be operational before the 2020s.

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 These and other nuclear power units now operating have been found to be safe and reliable, but they are being superseded by better designs.

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

So-called third-generation reactors have:

- A more 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.*
- Substantial grace period, so that following shutdown the plant requires no active intervention for (typically) 72 hours.
- Stronger reinforcement against aircraft impact than earlier designs, to resist radiological release.
- Higher burn-up to use fuel more fully and efficiently, and reduce the amount of waste.
- Greater use of burnable absorbers ('poisons') to extend fuel life.

* The US NRC requirement for calculated core damage frequency (CDF) 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 most designs now in operation 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 most will be designed for load-following. European Utility Requirements (EUR) since 2001 specify that new reactor designs must be capable of load-following between 50 and 100% of capacity. While most French reactors 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.

A feature of some new designs is modular construction. The means that many small components are assembled in a factory environment (offsite or onsite) into structural modules weighing up to 1000 tonnes, and these can be hoisted into place. Construction is speeded up.

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

However, certification of designs is on a national basis, and is safety-based – see section below.

Another feature of some new designs is modular construction. Large structural and mechanical sections of the plant of up to 1000 tonnes each are manufactured in factories or on site adjacent to the plant and lifted into place, potentially speeding construction.

A contrast between the 1188 MWe Westinghouse reactor at Sizewell B in the UK and the modern Westinghouse AP1000 of similar power illustrates the evolution from 1970-80 types. First, the AP1000 footprint is very much smaller – about one-quarter the size, secondly the concrete and steel requirements are lower by a factor of five*, and thirdly it has modular construction. A single unit has 149 structural modules broadly 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 offsite in parallel with the onsite construction.

* Sizewell B: 520,000 m^3 concrete (438 m^3/MWe), 65,000 t rebar (55 t/MWe);
AP1000: <100,000 m^3 concrete (90 m^3/MWe, <12,000 t rebar (11 t/MWe).

At Sanmen and Haiyang in China, where the first AP1000 units were grid connected in August 2018, the first module lifted into place weighed 840 tonnes. More than 50 other modules used in the reactors' construction weigh more than 100 tonnes, while 18 weigh in excess of 500 tonnes.

US, EU and UK 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 fell 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 was an advanced boiling water reactor (ABWR) derived from a General Electric design and then promoted both by GE Hitachi and Toshiba as a proven design, which is in service in Japan and was being built in Taiwan. Four are planned in the UK.

The other type, System 80+, was an advanced pressurised water reactor, which was ready for commercialisation but was never promoted for sale. It was the basis of the Korean Next Generation Reactor programme and many of its design features are incorporated into eight South Korean reactors, specifically the APR1400, which is operating in South Korea and being built in South Korea and the UAE and 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 and is undergoing the generic design assessment process in the UK (see below).

Another, more innovative US advanced reactor was smaller – 600 MWe – and had passive safety features (its projected core damage frequency is more than 100 times less than 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 were valid for 15 years. As a result of an exhaustive public process, safety issues within the scope of the certified designs were fully resolved and hence are not open to legal challenge during licensing for particular plants. Using such certified designs, US utilities are able to obtain a single NRC licence to both construct and operate a reactor before construction begins.

Both GE Hitachi and Toshiba in 2010 submitted separate applications to renew the US design certification for their respective versions of the ABWR (Toshiba's incorporating design changes already submitted to the NRC in connection with the South Texas Project combined construction and operating licence application). The Japanese version of it differs in allowing modular construction, so is not identical to that licensed in the USA. In mid-2016 Toshiba withdrew its design certification renewal application, and in August 2017 GE Hitachi put its review by the NRC on hold.

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 meant that prospective buyers then had 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 III+ 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 (GDA, pre-licensing approval) based on the NRC design certification, and expressing its policy of global standardisation. The application was supported by European utilities, and was granted in 2017.

Overnight capital costs were projected to be very competitive with older designs, 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 (four units under construction, with many more to follow) and in the USA (initially four units at two sites). It is planned for building in the UK. 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 was completed with revised design certification in December 2011. The NRC chairman said that the revised AP1000 design is one that seems to most fully meet the expectations of the commission’s policy statement on advanced reactors. "The design provides enhanced safety margins through use of simplified, inherent, passive or other innovative safety and security functions, and also has been assessed to ensure it could withstand damage from an aircraft impact without significant release of radioactive materials." This design change increased the capital cost.

In December 2016 Westinghouse requested the NRC to extend the design certification of its AP1000 reactor for five years from 2021 to 2026. In the light of operational experience of the first few reactors it would then apply for renewal of US design certification.

The **ESBWR** from GE Hitachi received US design certification in September 2014.

The South Korean **APR1400** received US design certification in August 2019.

In January 2017 **NuScale** submitted its small modular reactor design to the NRC for design certification. The application consisted of nearly 12,000 pages of technical information. The certification process is expected to take 40 months. See information paper on Small Nuclear Power Reactors for reactor details.

Longer term, the NRC expected to review 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. It will also focus on small reactor designs.

In **Europe** there are moves towards harmonised requirements for licensing. Here, since 1991, reactors may also be certified according to compliance with European Utility Requirements (EUR) of 12 generating companies, which have stringent safety criteria. The EUR are essentially a utilities' wish list of some 5000 items needed for new nuclear plants. Designs certified as complying with EUR include Westinghouse's AP1000, Gidropress's AES-92 and VVER-TOI, Areva's EPR, Mitsubishi’s EU-APWR and in 2017 KHNP's APR1400 (EU-APR). GE's ABWR, Areva's Kerena, and Westinghouse's BWR 90 also have some measure of EUR approval. China's Hualong One are under review.

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.

The UK’s Office for Nuclear Regulation (ONR) undertakes generic design assessment (GDA) of nuclear reactors. A GDA of each type can then be followed by site- and operator-specific licensing. ONR made initial assessments of four designs which were submitted in 2007: UK EPR for Areva, AP1000 for Westinghouse, ESBWR for GE Hitachi, and ACR-1000 for AECL in Canada. The latter two were withdrawn from the process in 2008 and in 2013 the GE Hitachi ABWR was added. The ONR and Environment Agency jointly issued design acceptance confirmations (DAC), and statements on design acceptability (SODA) for the EPR December 2012, and for the AP1000 in March 2017. In 2013 Hitachi-GE applied for UK generic design approval for the ABWR, and after some design changes this is likely to be granted at the end of 2017.

As the GDA for the EPR design proceeded, issues arose which were in common with new capacity being built elsewhere, particularly the EPR units in Finland and France. This led to international collaboration and a joint regulatory statement on the EPR instrumentation and control among ONR, US NRC, France's ASN and Finland's STUK. More broadly it relates to the Multinational Design Evaluation Programme and will help improve the harmonization of regulatory requirements internationally.

In 2012 Rosatom announced that it intended to apply for design certification for its VVER-TOI reactor design of 1200 MWe, with a view to Rusatom Overseas building them in UK.

In 2016 China General Nuclear Power Group (CGN) applied for GDA for the 1150 MWe Hualong One (HPR1000) reactor design, with a view to building it at Bradwell. General Nuclear Systems, a joint venture with EDF holding 33.5% and CGN 66.5%, was formed for progressing the GDA, which commenced in January 2017 and moved to its fourth and final stage in February 2020.

Small modular reactors (SMRs) are a further GDA task impending for the ONR.

Joint initiatives and collaboration

Three 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:

The Multinational Design Evaluation Program (MDEP) was launched in 2006 by the US NRC and the French Nuclear Safety Authority (ASN) to develop innovative approaches to leverage the resources and knowledge of national regulatory authorities reviewing new reactor designs. It is led by the OECD Nuclear Energy Agency and involves the IAEA. Ultimately it aims to develop multinational regulatory standards for design of Gen IV reactors. The US Nuclear Regulatory Commission (NRC) has proposed a three-stage process culminating in international design certification for new reactor types, notably Generation IV types. Twelve countries are involved so far: Canada, China, Finland, France, India (from 2012), Japan, Korea, Russia, South Africa, Sweden (from 2013), UK, USA, and others which have or are likely to have firm commitments to building new nuclear plants may be admitted – the UAE is an associate member.

The MDEP pools the resources of its member nuclear regulatory authorities for the purpose of: 1) co-operating on safety reviews of designs of nuclear reactors that are under construction and undergoing licensing in several countries; and 2) exploring opportunities and potential for harmonisation of regulatory requirements and practices. It also produces reports and guidance documents that are shared internationally beyond the MDEP membership.

The 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 Nuclear Reactors information paper.

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 in the world reactor supply market, and since then another has become prominent:

- 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), 60% GE; and Hitachi-GE Nuclear Energy based in Japan, 80% Hitachi.
- Westinghouse had become a 77%-owned subsidiary of Toshiba (with The Shaw Group 20%). Toshiba is now an 87% owner, having sold 10% to Kazatomprom and bought the 20% share.

Ten years later, in 2016, Westinghouse has collaborated with China's State Nuclear Power Technology Corporation (SNPTC) in developing the AP1000 design to a CAP1000 and also a larger CAP-1400, and China is gaining a high profile as reactor vendor alongside Russia's Rosatom. Areva was substantially restructured due to huge cost overruns on two EPR projects, and Electricite de France (EDF) took over the nuclear power plant part. Japanese vendors are overshadowed by the after-effects of the Fukushima accident. South Korea's KEPCO through KHNP is building its APR1400 on budget and schedule in the United Arab Emirates, but faces new political challenges at home.

There have also 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.

Who is marketing what?

Apart from small reactors, the following are the main models actively being marketed:

- EDF (Framatome): EPR2, Atmea1, Kerena
- Westinghouse: AP1000
- GE Hitachi: ABWR, ESBWR, PRISM
- KHNP: APR1400, EU-APR
- Mitsubishi: APWR, Atmea1
- Rosatom: AES-92, AES-2006, VVER-TOI
- SNC-Lavalin: EC6
- CNNC & CGN: Hualong One
- SNPTC: CAP1400

Advanced power reactors operational

Developer	Reactor	Size – MWe gross	Design progress, notes
GE Hitachi, Toshiba	ABWR	1380	Commercial operation in Japan since 1996-7. US design certification 1997. UK design certification application 2013. Active safety systems.
KHNP	APR1400 (PWR)	1450	Shin Kori 3&4 operating in South Korea. Under construction: Shin Hanul 1&2 in South Korea, Barakah in UAE. Korean design certification 2003. US design certification August 2019.
Gidropress	VVER-1200 (PWR)	1200	Novovoronezh II, from mid-2016, Leningrad II from 2018, as AES-2006. Under construction at Akkuyu in Turkey and Rooppur in Bangladesh.
OKBM	BN-800	880	Beloyarsk 4, demonstration fast reactor and test plant.
Westinghouse	AP1000 (PWR)	1250	Four units operating in China and under construction in the USA; many units planned in China (as CAP1000).
Areva (& EdF)	EPR (PWR)	1750	Two units operating in China, under construction in Finland and France.

Advanced power reactors under construction

Developer	Reactor	Size – MWe gross	Design progress, notes
Gidropress	VVER-TOI (PWR)	1300	Under construction at Kursk II, planned for Nizhny Novgorod and many more in Russia.
CNNC & CGN (China)	Hualong One (PWR)	1170	Main Chinese export design, under construction at Fangchenggang and Fuqing, also Pakistan.

INET & CNEC (China)	HTR-PM, HTR-200 module	2x105 (one module)	Demonstration plant being built at Shidaowan.

Advanced power reactors ready for deployment

Developer	Reactor	Size – MWe gross	Design progress, notes
GE Hitachi	ESBWR	1600	Planned for Fermi and North Anna in USA. Developed from ABWR, but passive safety systems. Design certification in USA Sept 2014.
Mitsubishi	APWR	1530	Planned for Tsuruga in Japan. US design certification application for US-APWR, but delayed. EU design approval for EU-APWR Oct 2014.
Areva & Mitsubishi	Atmea1 (PWR)	1150	Planned for Sinop in Turkey. French design approval Feb 2012. Canadian design certification in progress.
Candu Energy	EC6 (PHWR)	750	Improved CANDU-6 model. Canadian design certification June 2013.

Light water reactors

(Power reactors moderated and cooled by water)

EPR

Areva NP (formerly Framatome ANP) developed a large (4590 MWt, typically 1750 MWe gross and 1630 MWe net) European pressurised water reactor (EPR), which was accepted in mid-1995 as the new standard design for France and received French design approval in 2004. It is a four-loop design derived from the German Konvoi types with features from the French N4, and was expected to provide power about 10% cheaper than the N4. 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 double containment with four separate, redundant active safety systems, and boasts a core catcher under the pressure vessel. The safety systems are physically separated through four ancillary buildings on the same concrete raft, and two of them are aircraft crash protected. The primary diesel generators have fuel for 72 hours, the secondary back-up ones for 24 hours, and tertiary battery back-up lasts 12 hours. It is designed to withstand seismic ground acceleration of 600 Gal without safety impairment.

The first EPR unit commenced construction at Olkiluoto in Finland, the second at Flamanville in France, the third European one was to be at Penly in France. However the first EPR to be grid connected was at Taishan in China. It entered commercial operation at the end of 2018. The EPR has undergone UK generic design assessment, with some significant changes to instrumentation and control systems being agreed with other national regulators, and two are being built at Hinkley Point C in the UK.

Questions arose regarding the steel quality in the top and bottom reactor pressure vessel heads for Flamanville, forged by Areva's Creusot Forge plant. The pressure vessel for Olkiluoto was forged in Japan, and those for Taishan by MHI and Dongfang Electric.

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, but this process is suspended. The first unit (with 80% US content) was 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 was said to be making the necessary changes to output electricity at 60 Hz instead of the original design's 50 Hz. The main development of the type was to be through UniStar Nuclear Energy.

Areva NP is working with EdF on a 'new model' EPR, the EPR NM or **EPR2**, "offering the same characteristics" as the EPR but with simplified construction and significant cost reduction – about 30%. The basic design was to be completed in 2020, and in mid-2019 the French regulator ASN said it was happy with most aspects of the design. Emergency core cooling is significantly different to the EPR. EdF said that it, not the complex EPR being built at Flamanville, would be the model that replaced the French fleet from the late 2020s. Poland appears to be a candidate for the demonstration plant.

AP1000

The Westinghouse AP1000 is a two-loop PWR which has evolved from the smaller AP600, one of the first new reactor designs certified by the US NRC. 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. It has a core cooling system including passive residual heat removal by convection, improved containment isolation, passive containment cooling system to the atmosphere and in-vessel retention of core damage (corium) with water cooling around it. No safety-related pumps or ventilation systems are needed. The AP1000 gained US design certification in 2005, and UK generic design assessment approval in 2017. However, the structural design for the USA and UK was significantly modified from 2008 to withstand aircraft impact.

It has been built in China at Sanmen and Haiyang, and is under construction at Vogtle in the USA. The units are being assembled from modules. It is 1250 MWe gross and 1110-1117 MWe net in the USA, 1157 or 1170 MWe net in China (3415 MWt). Westinghouse earlier claimed a 36-month construction time to fuel loading. The first ones being built in China were on a 57-month schedule to grid connection, but took about 110 months. Progress was delayed, particularly by the need to re-engineer the 91-tonne coolant pumps, of which each rector has four. After the first four units in China, the design is known as the CAP1000 there.

CAP1400

Westinghouse has been working with SNPTC and SNERDI in China to develop jointly a passively safe 1500 MWe (4040 MWt) two-loop design from the AP1000, the CAP1400, with 193 fuel assemblies and improved steam generators, operating at 323°C outlet temperature, 60-year design lifetime, and 72-hour non-intervention period in event of accident. Average discharge burn-up is about 50 GWd/t, maximum 59.5 GWd/t. Operation flexibility includes extra control rods for MOX capability, 18 to 24-month cycle, and load-following. Seismic rating is 300 gal. The CAP1400 project may extend to a larger, three-loop CAP1700 or CAP2100 design if the passive cooling system can be scaled to that level. Westinghouse has agreed that SNPTC will own the intellectual property rights for any AP1000 derivatives over 1350 MWe. Construction of the first unit at Shidaowan is expected to start about the end of 2017.

ABWR

The advanced boiling water reactor (ABWR) is derived from a General Electric design in collaboration with Toshiba. Two examples built by Hitachi and two by Toshiba have been in commercial operation in Japan (1315 MWe net), with another two under construction there and two in Taiwan. More are planned in Japan and four are planned in the UK.

The ABWR has been offered in slightly different versions by GE Hitachi, Hitachi-GE and Toshiba, so that 'ABWR' is now a generic term. It is basically a 1380 MWe (gross) unit (3926 MWt in Toshiba version), though GE Hitachi quote 1350-1600 MWe net. Toshiba outlines development from its 1400 MWe class to a 1500-1600 MWe class unit (4300 MWt). Tepco was funding the design of a next generation BWR, and the ABWR-II is quoted as 1717 MWe.

Toshiba was promoting its EU-ABWR of 1600 MWe with core catcher and filtered vent, developed with Westinghouse Sweden. The Hitachi UK-ABWR may have similar features but be similar size to Japanese units.

The first four ABWRs were each built in 39-43 months on a single-shift basis. Though GE and Hitachi have subsequently joined up, Toshiba retains some rights over the design, as does Tepco. The design can run on full-core mixed oxide (MOX) fuel, as for the Ohma plant being built in Japan. Design life is 60 years. Unlike previous BWRs in Japan the external recirculation loop and internal jet pumps are replaced by coolant pumps mounted at the bottom of the reactor pressure vessel. Safety systems are active – GEH describes it as “the pinnacle of the evolution of active safety.”

Both Toshiba and GE Hitachi have applied separately to the NRC for design certification renewal, though these are respectively withdrawn or on hold. The initial certification in 1997 was for 15 years and in 2011 the NRC certified for GE Hitachi an evolved version which allows for aircraft impacts. UK generic design assessment approval for Hitachi's version of the ABWR is expected at the end of 2017.

GE Hitachi was also designing a 600-800 MWe version of the ABWR, with five instead of ten internal coolant pumps, aiming at Southeast Asia. In addition, a 400 MWe version was envisaged.

ESBWR

GE Hitachi Nuclear Energy's **ESBWR** is an improved design "evolved from the ABWR" but that utilizes passive safety features including natural circulation principles. It was developed from a predecessor design, the SBWR at 670 MWe. GEH says it is safer and more efficient than earlier models, with 25% fewer pumps, valves and motors, and can maintain cooling for seven days after shutdown with no AC or battery power. The emergency core cooling system has eliminated the need for pumps, using passive and stored energy. The used fuel pool is below ground level.

The ESBWR (4500 MWt) will produce approximately 1600 MWe gross, and 1520 MWe net, depending on site conditions, and has a design life of 60 years. It is more fully known as the Economic Simplified BWR (ESBWR) and leverages proven technologies from the ABWR. GE Hitachi gained US NRC design certification for the ESBWR in September 2014, following design approval in March 2011. It was submitted for UK generic design assessment in 2007, but withdrawn a year later.

GEH is selling this alongside the ABWR, which it characterises as more expensive to build and operate, but proven. The ESBWR is more innovative, with lower building costs due to modular construction, lower operating costs, 24-month refuelling cycle and a 60-year operating lifetime. In the USA plans to build as Detroit Edison's Fermi 3 and Dominion's North Anna 3 are not proceeding.

APWR

Mitsubishi's large APWR – advanced PWR of 1538 MWe gross (4451 or 4466 MWt) – was developed in collaboration with four utilities (Westinghouse was earlier involved). The first two are planned for Tsuruga, originally to come online from 2016. It is a four-loop design with 257 fuel assemblies and neutron reflector, is simpler, combines active and passive cooling systems in a double containment, and has over 55 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** is 4451 MWt, about 1600 MWe net, due to longer (4.3m instead of 3.7m) fuel assemblies, higher burn-up (62 GWd/t) and higher thermal efficiency (37%) (2013 company description). It has 24-month refuelling cycle. Its emergency core cooling system (ECCS) has four independent trains, and its outer walls and roof are 1.8 m thick. US design certification application was in January 2008 with certification expected in 2016, but halted. In March 2008 MHI submitted the same design for EUR (European Utility Requirements) certification, as the **EU-APWR,** and this certification of compliance was granted in October 2014. MHI planned to join with Iberdrola Engineering & Construction in bidding for sales of this in Europe. Iberdrola would be responsible for building the plants.

The Japanese government was expected to provide financial support for US licensing of the US-APWR. Washington Group International was to be involved in US developments with Mitsubishi Heavy Industries (MHI). The US-APWR was selected by Luminant for Comanche Peak, Texas, a merchant plant.

APR1400, EU-APR, APR+, APR1000

South Korea's APR1400 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 in Korean conditions according to an IAEA status report, 1350-1400 MWe net (3983 - nominal 4000 MWt) with two-loop primary circuit. The first of these are operating in Korea – Shin Kori 3&4 – with Shin Hanul 1&2 under construction. It was chosen for the United Arab Emirates (UAE) nuclear programme on the basis of cost and reliable building schedule, and four units are under construction there, with the first expected online in 2020.

Fuel in 241 fuel assemblies has burnable poison and will have up to 55 GWd/t burn-up, refuelling cycle around 18 months, outlet temperature 324ºC. It is designed "not only for the base-load full power operation but also for a part load operation such as the load following operation. A standard 100-50-100% daily load follow operation has been considered in the reactor core design as well as in the plant control systems." Ramp up and down between 100% and 50% takes two hours. Plant operating lifetime is 60 years, seismic design basis is 300 Gal. A low-speed (1800 rpm) turbine is used. An application for US design certification was lodged in 2013 and a revised version accepted in March 2015. The NRC confirmed its safety in September 2018 and design certification was approved in May 2019 and formally awarded in August.

Based on this, KOPEC has developed an EU version (APR1400-EUR or EU-APR) with double containment and core-catcher which was given EUR approval in October 2017. It is 4000 MWt, 1520 MWe gross, with a design lifetime of 60 years and 250 Gal seismic rating.

KHNP is also developing a more advanced 4308 MWt, 1560 MWe (gross) version of the APR1400, the **APR+**, which gained design approval from NSSC in August 2014. It was "developed with original domestic technology", up to 100% localized, over seven years since 2007, with export markets in view. It has modular construction which is expected to give 36-month construction time instead of 52 months for the APR1400. It has 257 fuel assemblies of a new design, 18- to 24-month fuel cycle, and passive decay heat removal. Also it is more highly reinforced against aircraft impact than any earlier designs. Seismic rating is 300 Gal.

In addition some of the APR features are being incorporated into an exportable **APR-1000** intended for overseas markets, notably Middle East and Southeast Asia, and will be able to operate with an ultimate heat sink of 40°C, instead of 35°C for the OPR-1000. Improved safety and performance will raise the capital cost above that of the OPR, but it this will be offset by reduced construction time (40 months instead of 46) due to modular construction.

Atmea1

The **Atmea1** has been developed by the Atmea joint venture established in 2007 by Areva NP and Mitsubishi Heavy Industries to produce an evolutionary 1100-1150 MWe net (3150 MWt) three-loop PWR using the same steam generators as EPR. This has 37% net thermal efficiency, 157 fuel assemblies 4.2 m long, 60-year operating lifetime, and the capacity to use mixed-oxide fuel for full core load. Fuel cycle is flexible 12 to 24 months with short refuelling outage and the reactor has load-following (100-25% range) and frequency control capability. The first units are likely to be built at Sinop in Turkey.

Following an 18-month review, the French regulator ASN approved the general design in February 2012. The reactor is regarded as mid-sized relative to other modern designs and will be marketed primarily to countries embarking upon nuclear power programs. It has three active and passive redundant safety systems and an additional backup cooling chain, similar to EPR. It has a core-catcher, and is available for high-seismic sites. Canadian design certification is under way.

Kerena

Together with German utilities and safety authorities, Areva NP has also developed 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. It has not yet been submitted for certification anywhere, but is otherwise ready for commercial deployment.

It has two redundant active safety systems and two passive safety systems, including a core-catcher, similar to EPR. The reactor is simpler overall and uses high-burnup fuels (to 65 GWd/t) enriched to 3.54%, giving it refuelling intervals of up to 24 months. It can take a 50% MOX load, and uses flow variation to improve fuel usage. It has 37% net efficiency and can load-follow down to 70% using recirculation pumps only, and down to 40% with control rods.

AES-92, V-392

Gidropress late-model VVER-1000 units with enhanced safety (AES-92 & -91 power plants) have been built in India and China. Two more (V466B variant) were planned for Belene in Bulgaria. The **AES-92** is certified as meeting EUR, and its V-392 reactor is considered state of the art. They have four coolant loops, 163 fuel assemblies, and are rated 3000 MWt.

AES-2006, MIR-1200

A third-generation standardised **VVER-1200** (V-392M and V-491) reactor of 1198 MWe gross (with cool water) and 3212 MWt is in the AES-2006 plant. It is an evolutionary development of the well-proven VVER-1000 in the AES-92 and AES-91 plants, with longer life (60 years for non-replaceable equipment), greater power, and greater efficiency (34.8% net instead of 31.6%) and 60 GWd/t burn-up. Cogeneration heat supply capacity is 300 MWt. It retains four coolant loops and has 163 FA-2 fuel assemblies, each with 534 kg of UO_2 fuel enriched to 4.95%. Core outlet temperature is 329°C.

The lead units are being built at Novovoronezh II (V-392M) and Leningrad II (V-491), the first one starting operation in 2016. The Novovoronezh units provide 1114 MWe net each, and the Leningrad II units 1085 MWe net each. Two steam turbines are offered: Power Machines (Silmash) full-speed; and Alstom Arabelle half-speed, as proposed for MIR-1200 and Hanhikivi in Finland.

An AES-2006 plant will consist of two of these OKB Gidropress reactor units expected to run for 60 years with capacity factor of 90%. Overnight capital cost was said to be US$ 1200/kW (though the first contract was about $2100/kW) and serial construction time 54 months. They have enhanced safety including that related to earthquakes and aircraft impact (V-392M especially) with some passive safety features, double containment, and core-catcher. Planned for Akkuyu in Turkey (V-509).

While Gidropress is responsible for the actual 1200 MWe reactor, Moscow AEP and Atomproekt St Petersburg are going different ways on the cooling systems, and the V-392M version is the basis of the VVER-TOI. Passive safety systems prevail in Moscow's V-392M design, while St Petersburg's V-491 design focuses on active safety systems based on the Tianwan V-428 design. In both, long-term decay heat removal does not rely on electrical power or ultimate heat sink. (Details in the information paper on Nuclear Power in Russia.) Atomenergoproekt says that the AES-2006 conforms to both Russian standards and European Utilities Requirements (EUR). In Europe the V-491 technology is being called the Europe-tailored reactor design, **MIR-1200** (Modernised International Reactor) or AES-2006E, with some Czech involvement. Those bid for Temelin are quoted as 1158 MWe gross,1078 MWe net. That for Hanhikivi is 1250 MWe gross, due to cold water.

VVER-TOI

In 2010 Atomenergoproekt announced the **VVER-TOI** (typical optimised, with enhanced information) design based on V-392M. The basic Gidropress reactor is V-510. It has upgraded pressure vessel, increased power to 3300 MWt and 1255 MWe gross (nominally 1300, hence VVER-1300), improved core design still with 163 fuel assemblies to increase cooling reliability, larger steam generators, further development of passive safety with 72-hour grace period requiring no operator intervention after shutdown, lower construction and operating costs, and 40-month construction time. It will use a low-speed turbine-generator and can undertake daily load-following down to 50% of power. The project was initiated in 2009 and the design was completed at the end of 2012. In June 2012 Rosatom said it would apply for design certification in UK through Rusatom Overseas, with the VVER-TOI version. The first units are planned for Kursk II and Nizhny Novgorod in Russia.

Details of MIR-1200 and VVER-TOI are in the Nuclear Power in Russia information paper.

VVER-600

Gidropress has developed the VVER-600/V-498 for sites such as Kola, where larger units are not required. It is a two-loop design based on the V-491 St Petersburg version of the VVER-1200 and using the same basic equipment but without core-catcher (corium retained within RPV). It will have 60-year life and is capable of load-following. Export potential is anticipated. It supercedes the VVER-640/V-407 design.

Hualong One, HPR1000

In China, there are two indigenous designs based on a French predecessor but developed with modern features. CNNC developed the ACP1000 design, with 1100 MWe nominal power and load-following capability, and 177 fuel assemblies. In parallel but somewhat ahead, China Guangdong Nuclear Power Corporation, now China General Nuclear Power (CGN) led the development of the 1100 MWe ACPR-1000, with 157 fuel assemblies (same as the French M-310 predecessor), and about 30 of these have been built. However, due to rationalisation over 2011-13, this design has been dropped in favour of the Hualong One, essentially the ACP1000 with some features from the ACPR.

The Hualong One thus has 177 fuel assemblies 3.66 m long, 18-24 month refuelling interval. It has three coolant loops delivering 3050 MWt, 1170 MWe gross, 1090 MWe net (CNNC version). It has double containment and active safety systems with some passive elements, and a 60-year design lifetime. Average burnup is 45,000 MWd/tU, thermal efficiency is 36%. Seismic shutdown is at 300 gal. Instrumentation and control systems will be from Areva-Siemens. Estimated cost in China is $3500/kWe. The first units under construction are Fangchenggang 3&4 (CGN) and Fuqing 5&6 (CNNC). It is also being built in Pakistan.

CNNC and CGN in December 2015 formed a 50-50 joint venture company – Hualong International Nuclear Power Technology Co – to market it. The version promoted on the international market, is called HPR1000 (Hualong Pressurized Reactor 1000), based on the CGN version, with Fangchenggang as the reference plant. In October 2015 CGN submitted the HPR1000 for certification of compliance with European Utility Requirements (EUR).

Fuller details of the situation are in the Nuclear Power in China information paper.

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 information paper.

Other advanced PWR ventures and concepts are in Appendix 2.

Heavy water reactors

(Moderated and mostly cooled by heavy water)

In Canada, the government-owned Atomic Energy of Canada Ltd (AECL) had two designs under development which are based on its reliable CANDU-6 reactors, the most recent of which are operating in China. In 2011 the reactor division of AECL was sold and became Candu Energy Inc, a subsidiary of SNC-Lavalin. One of these earlier designs continues, with associated fuel cycle innovation.

The CANDU-9 (925-1300 MWe) was developed from the CANDU-6 also as a single-unit plant. It had flexible fuel requirements which have been taken forward to the EC6. 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 the CANDU-9, along with experience in building recent Korean and Chinese units, was then put back into the Enhanced CANDU-6 (EC6). This is to be built as twin units – with power increase to 740-750 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). EC6 is presented as a third-generation design based on Qinshan Phase III in China, and is under consideration for new build in Ontario and overseas. Phase 2 of CNSC's vendor pre-project design review was completed in April 2012, with phase 3 on target for 2013.

Versatility of fuel is a claimed feature of the EC6 and its derivatives. As well as natural uranium, it can use direct recovered/reprocessed uranium (RU) from used PWR fuel, natural uranium equivalent (NUE – DU + RU), MOX (DU + Pu), fertile fuels such as LEU + thorium and Th with Pu, and closed cycle fuels (Th + U-233 + Pu). The NUE fuel cycle with full-core NUE is being demonstrated at Qinshan in China in CANDU-6 units*. There is also a program for the Advanced Fuel Candu Reactor (AFCR) – an adaptation of EC6 – on direct use of RU, and also LEU + thorium-based CANDU fuel. Finally a CANMOX fuel is proposed with EC6 for disposal of the UK's plutonium stock.

* RU with 0.9% U-235 plus DU gives 0.7% NUE, which is burned down to about 0.25% U-235.

The EC6 has design features, notably its automated refuelling, which enable third-party process monitoring in relation to non-proliferation concerns.

AFCR

The Advanced Fuel CANDU Reactor (AFCR) is a 740 MWe development of the EC6, designed to use recycled uranium and also thorium-based fuels. It has been developed by Candu Energy with CNNC's Third Qinshan Nuclear Power Corp, which plans to convert the two Qinshan CANDU-6 PHWR units to AFCRs. Then new-build AFCRs are envisaged in China. One AFCR can be fully fuelled by the recycled uranium from four LWRs' used fuel. Hence deployment of AFCRs will greatly reduce the task of managing used fuel and disposing of high-level waste, and could reduce China's fresh uranium requirements. Late in 2014 a joint venture framework agreement between CNNC and Candu Energy was signed to build AFCR projects domestically and develop opportunities for them internationally. In September 2016 an agreement among SNC-Lavalin, CNNC and Shanghai Electric Group was to set up a joint venture in mid-2017 to develop, market and build the AFCR, with NUE fuel.

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 pressurised light water boiling at 285ºC and 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. A co-located fuel cycle facility is planned, with remote handling for the highly-radioactive fresh fuel. At the end of 2016 the design was complete and large-scale engineering studies were validating innovative features of the design. No site or construction schedule had been announced for the demonstration unit.

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

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.

Other advanced PHWR designs and concepts are in Appendix 3.

High-temperature gas-cooled reactors

(Graphite-moderated)

These reactors use helium as a coolant at up to 950ºC, which either makes steam conventionally (Rankine cycle) or directly drives a gas turbine for electricity and a compressor to return the gas to the reactor core (Brayton cycle). 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, HTR-PM 600

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 China Nuclear Engineering & Construction Group (CNEC), with China Huaneng Group leading the demonstration plant project. This will have two reactor modules, each of 250 MWt/105 MWe (equivalent), with a single steam generator, and using 8.5% enriched fuel (245,000 elements) giving 90 GWd/t discharge burnup. With an outlet temperature of 750ºC the pair will produce steam at 566ºC to drive a single steam cycle turbine at about 40% thermal efficiency.

This 210 MWe Shidaowan demonstration plant is to pave the way for commercial 600 MWe reactor units using the twin reactor modules (3x210 MWe), also using the steam cycle. These are being promoted by CNEC. Plant life is envisaged as 40 years with 85% load factor.

Other advanced HTR designs and concepts are in Appendix 4.

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

Fast neutron reactors

(Not moderated, cooled by liquid metal)

Fuller description of fast neutron reactors is in that paper.

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 (*i.e.* 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, and can readily use the world's 1.5 million tonnes of depleted uranium as fuel. They are now often designed to burn actinides as well.

About 20 liquid metal-cooled FBRs have already been operating, some since the 1950s, and some have supplied electricity commercially. About 400 reactor-years of operating experience have been accumulated. Today Russia and India have FNRs high profile in their nuclear programs, with Japan, China and France also significant. See also Fast Neutron Reactors paper.

India's 500 MWe prototype fast breeder reactor at Kalpakkam is expected to be operating in 2018, 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.

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 core is 0.88 metres active height and 0.75 m diameter. The BN-350 FBR operated in Kazakhstan for 27 years and about half of its output was used for water desalination. The BN-600 is configured to burn the plutonium from its military stockpiles.

BN-800

The first (and probably only Russian) BN-800, a new more powerful (789 MWe, 880 MWe gross, 2100 MWt) fast neutron reactor from OKBM with Atomproekt at St Petersburg with improved features, was grid-connected at Beloyarsk in December 2015. It is designed to have considerable fuel flexibility – U+Pu nitride, MOX, or metal, and with breeding ratio up to 1.3, though only 1.0 as configured at Beloyarsk. The core is a similar size to that of the BN-600. Initially it is being run with one-fifth MOX fuel, but will have a full MOX core from about 2020. It does not have a breeding blanket, though a version designed for Sanming in China has up to 198 DU fuel elements in a blanket. Its main purpose is to provide operating experience and technological solutions, especially regarding fuels, that will be applied to the BN-1200. Further details in the information paper on Fast Neutron Reactors.

BN-1200

The BN-1200 is being designed by OKBM for operation with MOX fuel initially and dense nitride U-Pu fuel subsequently, in closed fuel cycle. It is significantly different from preceding BN models, and Rosatom plans to submit the BN-1200 to the Generation IV International Forum (GIF) as a Generation IV design. The BN-1200 has a capacity of 2900 MWt (1220 MWe gross), a 60-year design life, and burn-up of up to 120 GWd/t. The capital cost is expected to be much the same as that of the VVER-1200. Its breeding ratio is quoted as 1.2 to 1.4, using oxide or nitride fuel. OKBM envisages about 11 GWe of such plants by 2030, including South Urals nuclear plant. The detailed design was completed in May 2017, and the first unit is to be built at Beloyarsk possibly from 2020. This is part of a federal Rosatom program, the Proryv (Breakthrough) Project for large fast neutron reactors.

BREST

Russia has experimented with several lead-cooled reactor designs, and used lead-bismuth cooling for 40 years in reactors for its seven Alfa class submarines. Pb-208 (54% of naturally-occurring lead) is transparent to neutrons. A significant new Russian design from NIKIET is the BREST-300 fast neutron reactor, of 300 MWe (700 MWt) with lead as the primary coolant, at 540ºC, 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. Used fuel can be recycled indefinitely, with on-site reprocessing and associated facilities. A demonstration unit is planned at Seversk by 2022, and 1200 MWe (2800 MWt) units are proposed. Both designs have two cooling loops. BREST-300 has 17.6 tonnes of fuel, BREST-1200 about 60 tonnes. See information paper on Nuclear Power in Russia for further details.

PRISM

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. Each PRISM Power Block consists of two modules of 840 MWt, 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. PRISM is suited to operation with dry cooling towers due to high thermal efficiency and small size.

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. Breeding ratio depends on purpose and hence configuration, so ranges from 0.72 for used LWR recycle to 1.23 for breeder. Used PRISM fuel is recycled after removal of fission products. The commercial-scale plant concept, part of an 'Advanced Recycling Center', uses three power blocks (six reactor modules) to provide 1866 MWe. See also *Electrometallurgical 'pyroprocessing'* section in Processing Used Nuclear Fuel information paper.

A variant of this is proposed to utilise the UK's reactor-grade plutonium stockpile. A pair of PRISM units built at Sellafield would be operated initially so as to bring the material up to the highly-radioactive 'spent fuel standard' of self-protection and proliferation resistance. The whole stockpile could be irradiated thus in five years, with some by-product electricity and the plant would then proceed to re-use that stored fuel over perhaps 55 years solely for 600 MWe of electricity generation. GEH has launched a web portal in support of its proposal.

Westinghouse LFR

Westinghouse is developing a lead-cooled fast reactor (LFR) design with flexible output to complement intermittent renewable feed to the grid. Its high temperature capabilities will allow industrial heat applications. Westinghouse expects it to be very competitive, having low capital and construction costs with enhanced safety. Further operational and safety enhancements are also achieved by adoption of a fuel/cladding combination with high temperature capability based on those under development by Westinghouse in the Accident Tolerant Fuel program.

Japan

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 before closing down again due to an ancillary mechanical problem and is now being decommissioned. Mitsubishi Heavy Industries (MHI) is involved with a consortium to develop a Japan Standard Fast Reactor (JSFR) concept, though with breeding ratio less than 1:1. This is a large unit which would burn actinides with uranium and plutonium in oxide fuel. It could be of any size from 500 to 1500 MWe.

See also information paper on Fast Neutron Reactors.

Generation IV designs

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

Small reactors

See also information paper on Small Nuclear Power Reactors for other advanced designs, mostly under 300 MWe. This paper includes some designs which have become significantly larger than 300 MWe since first being described, but which are outside the mainstream categories dealt with here.

Accelerator-driven systems (ADS)

A related 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 subcritical, 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.

Notes & references

General sources

Nuclear Engineering International, various, and 2002 Reactor Design supplement. March 2012: Atmea1 reactor.
ABB Atom Dec 1999; Nukem market report July 2000;
The New Nuclear Power, 21st Century, Spring 2001,
Lauret, P. et al, 2001, The Nuclear Engineer 42, 5.
Smirnov V.S. et al, 2001, Design features of BREST reactors, KAIF/KNS conf.Proc.
OECD NEA 2001, Trends in the Nuclear Fuel Cycle
Technical and Economic Aspects of Load Following with Nuclear Power Plants, OECD Nuclear Energy Agency (June 2011)
Carroll D & Boardman C, 2002, The Super-PRISM Reactor System, The Nuclear Engineer 43,6;
Twilley R C 2002, Framatome ANP's SWR1000 reactor design, Nuclear News, Sept 2002.
Torgerson D F 2002, The ACR-700, Nuclear News Oct 2002.
IEA-NEA-IAEA 2002, Innovative Nuclear Reactor Development
Perera, J, 2003, Developing a passive heavy water reactor, Nuclear Engineering International, March
Sinha R.K.& Kakodkar A. 2003, Advanced Heavy Water Reactor, INS News vol 16, 1
US Dept of Energy, EIA 2003, New Reactor Designs.
Matzie R.A. 2003, PBMR - the first Generation IV reactor to be constructed, WNA Symposium
LaBar M. 2003, Status of the GT-MHR for electricity production, WNA Symposium
Carelli M 2003, IRIS: a global approach to nuclear power renaissance, Nuclear News Sept 2003
Perera J. 2004, Fuelling Innovation, IAEA Bulletin 46/1
AECL Candu-6 & ACR publicity, late 2005
IAEA Status report 83 – APR1400
Atomenergoproekt website

Appendices

Appendix 1: US Nuclear Regulatory Commission draft policy, May 2008

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.

Appendix 2: Other advanced PWR ventures and concepts

RMWR, RBWR

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 a 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 has also been closely involved, with its RBWR concept which has a major aim of burning actinides.

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.

Hitachi conducted joint research on RBWRs with MIT, University of Michigan, and UC Berkeley from 2007 to 2011, on the burning of transuranic elements. In a further stage of joint research from 2014, and applying the more accurate analysis methods developed by the three American universities, Hitachi will continue to evaluate the safety and performance of the new reactor concepts, and will study plans for tests with a view towards practical applications.

Norway's Thor Energy is exploring the operation of U-233 - thorium oxide (Th-MOX) fuel in an advanced reduced-moderation BWR (RBWR). This reactor platform, designed by Hitachi Ltd and JAEA, should be well-suited for achieving high U-233 conversion factors from thorium due to its epithermal neutron spectrum and flexible uranium-plutonium fuels in which high conversion or actinide destruction can be achieved. It is based on the ABWR architecture but has a shorter, flatter pancake-shaped core and a tight lattice to ensure sufficient fast neutron leakage and a negative void reactivity coefficient.

Areva-EdF-CGNPC project

Early in 2012 Areva and EdF agreed in principle with China Guangdong Nuclear Power group (CGN) to develop a mid-size PWR on the basis of CGNPC's CPR-1000, with third-generation safety features. A further three-way agreement was signed in September, with a view to having an outcome by mid-2013. It is not clear whether Mitsubishi Heavy Industries might be involved, though Areva has said that it wants the design "to have the highest possible technical convergence" with Atmea1. If a new reactor design results, it would be a competitor for Atmea1. However, Areva says that the talks are not aimed at joint development of a 1000 MWe reactor, so much as "to see if the three companies can converge on specifications for such a design that would allow deeper collaboration". This appears to have been overtaken by Hualong One.

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 third 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, *e.g.* 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 eight-year cycle, or equivalent MOX core. The core has low power density. US design certification was at pre-application review stage, but the concept appears to have evolved into the Westinghouse SMR. Estonia once expressed interest in building a pair of IRIS. Some consortium partners were interested in desalination, one in district heating.

The **VVER-1500** model was being developed by Gidropress. It will have enhanced safety, giving 1500 MWe gross from 4250 MWt. Design was expected to be complete in 2007 but the project was shelved in 2006 in favour of the evolutionary VVER-1200. It remains a four-loop design, with increased pressure vessel diameter to 5 metres, 241 fuel assemblies in core enriched to 4.4%, burn-up 45-55 and up to 60 GWd/t and life of 60 years. If revived, it will meet EUR criteria.

Appendix 3: Other advanced PHWR designs and concepts

ACR

The Advanced Candu Reactor (ACR), a third generation reactor design, was a more innovative concept, but has now been shelved. 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.

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) became the focus of attention by AECL (now Candu Energy Inc). 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-1000 was moving towards design certification in Canada, and a three-phase vendor pre-project design review was completed in 2010. In 2007 AECL applied for UK generic design assessment (pre-licensing approval) but then withdrew after the first stage. All licensing progress has ceased.

The **CANDU X** or SCWR is a variant of the ACR, but with supercritical light water coolant (*e.g.* 25 MPa and 625°C) 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.

The **Advanced Fuel CANDU Reactor** (AFCR) is being developed in China as a Generation III 700 MWe class reactor which essentially runs on the used fuel from four PWRs.

Appendix 4: Other advanced HTR designs and concepts

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 drew on German expertise and aimed for a step change in safety, economics and proliferation resistance. Production units would be 165 MWe. The PBMR would 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 would transfer heat in a steam generator rather than driving a turbine directly.) 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. In February 2010 General Atomics announced its Energy Multiplier Module (EM^2) design, superseding the GT-MHR.