Small Nuclear Power Reactors

(Updated January 2023)

- **There is strong interest in small and simpler units for generating electricity from nuclear power, and for process heat.**
- **This interest in small and medium nuclear power reactors is driven both by a desire to reduce the impact of capital costs and to provide power away from large grid systems.**
- **The technologies involved are numerous and very diverse.**

As nuclear power generation has become established since the 1950s, the size of reactor units has grown from 60 MWe to more than 1600 MWe, with corresponding economies of scale in operation. At the same time there have been many hundreds of smaller power reactors built for naval use (up to 190 MW thermal) and as neutron sources[a], yielding enormous expertise in the engineering of small power units and accumulating over 12,000 reactor years of experience.

The International Atomic Energy Agency (IAEA) defines 'small' as under 300 MWe, and up to about 700 MWe as 'medium' – including many operational units from the 20th century. Together they have been referred to by the IAEA as small and medium reactors (SMRs). However, 'SMR' is used more commonly as an acronym for 'small modular reactor', designed for serial construction and collectively to comprise a large nuclear power plant. (In this information page the use of diverse pre-fabricated modules to expedite the construction of a single large reactor is not relevant.) A subcategory of very small reactors – vSMRs – is proposed for units under about 15 MWe, especially for remote communities.

Small modular reactors (SMRs) are defined as nuclear reactors generally 300 MWe equivalent or less, designed with modular technology using module factory fabrication, pursuing economies of series production and short construction times. This definition, from the World Nuclear Association, is closely based on those from the IAEA and the US Nuclear Energy Institute. Some of the already-operating small reactors mentioned or tabulated below do not fit this definition, but most of those described do fit it. PWR types may have integral steam generators, in which case the reactor pressure vessel needs to be larger, limiting portability from factory to site. Hence many larger PWRs such as the Rolls-Royce UK SMR have external steam generators.

This information page focuses on advanced designs in the small category, *i.e.* those now being built for the first time or still on the drawing board, and some larger ones which are outside the mainstream categories dealt with in the Advanced Nuclear Power Reactors page. Some of the designs described here are not yet actually taking shape, others are operating or under construction. Four main options are being pursued: light water reactors, fast neutron reactors, graphite-moderated high temperature reactors and various kinds of molten salt reactors (MSRs). The first has the lowest technological risk, but the second (FNR) can be smaller, simpler and with longer operation before refuelling. Some MSRs are fast-spectrum.

Today, due partly to the high capital cost of large power reactors generating electricity via the steam cycle and partly to the need to service small electricity grids under about 4 GWe,[b] there is a move to develop smaller units. These may be built independently or as modules in a larger complex, with capacity added incrementally as required (see section below on Modular construction using small reactor units). Economies of scale are envisaged due to the numbers produced. There are also moves to develop independent small units for remote sites. Small units are seen as a much more manageable investment than big ones whose cost often rivals the capitalization of the utilities concerned.

An additional reason for interest in SMRs is that they can more readily slot into brownfield sites in place of decommissioned coal-fired plants, the units of which are seldom very large – more than 90% are under 500 MWe, and some are under 50 MWe. In the USA coal-fired units retired over 2010-12 averaged 97 MWe, and those expected to retire over 2015-25 average 145 MWe.

SMR development is proceeding in Western countries with a lot of private investment, including small companies. The involvement of these new investors indicates a profound shift taking place from government-led and -funded nuclear R&D to that led by the private sector and people with strong entrepreneurial goals, often linked to a social purpose. That purpose is often deployment of affordable clean energy, without carbon dioxide emissions.

A 2011 report for the US Department of Energy by the University of Chicago Energy Policy Institute[18] said that small reactors could significantly mitigate the financial risk associated with full-scale plants, potentially allowing small reactors to compete effectively with other energy sources.

Generally, modern small reactors for power generation, and especially SMRs, are expected to have greater simplicity of design, economy of series production largely in factories, short construction times, and reduced siting costs. Most are also designed for a high level of passive or inherent safety in the event of malfunction[c]. Also many are designed to be emplaced below ground level, giving a high resistance to terrorist threats. A 2010 report by a special committee convened by the American Nuclear Society showed that many safety provisions necessary, or at least prudent, in large reactors are not necessary in the small designs forthcoming. This is largely due to their higher surface area to volume (and core heat) ratio compared with large units. It means that a lot of the engineering for safety including heat removal in large reactors is not needed in the small reactors[d]. Since small reactors are envisaged as replacing fossil fuel plants in many situations, the emergency planning zone required is designed to be no more than about 300 m radius. The combined tables from this report are appended, along with notes of some early small water-, gas-, and liquid metal-cooled reactors.

Licensing is potentially a challenge for SMRs, as design certification, construction and operation licence costs are not necessarily less than for large reactors. Several developers have engaged with the Canadian Nuclear Safety Commission's (CNSC's) pre-licensing vendor design review process, which identifies fundamental barriers to licensing a new design in Canada and assures that a resolution path exists. The pre-licensing review is essentially a technical discussion, phase 1 of which involves about 5000 hours of staff time, considering the conceptual design and charged to the developer. Phase 2 is twice that, addressing system-level design.

A World Nuclear Association 2015 report on SMR standardization of licensing and harmonization of regulatory requirements[17] said that the enormous potential of SMRs rests on a number of factors:

- Because of their small size and modularity, SMRs could almost be completely built in a controlled factory setting and installed module by module, improving the level of construction quality and efficiency.
- Their small size and passive safety features lend them to countries with smaller grids and less experience of nuclear power.
- Size, construction efficiency and passive safety systems (requiring less redundancy) can lead to easier financing compared to that for larger plants.
- Moreover, achieving 'economies of series production' for a specific SMR design will reduce costs further.

The World Nuclear Association lists the features of an SMR, including:

- Small power and compact architecture and usually (at least for nuclear steam supply system and associated safety systems) employment of passive concepts. Therefore there is less reliance on active safety systems and additional pumps, as well as AC power for accident mitigation.
- The compact architecture enables modularity of fabrication (in-factory), which can also facilitate implementation of higher quality standards.
- Lower power leading to reduction of the source term as well as smaller radioactive inventory in a reactor (smaller reactors).
- Potential for sub-grade (underground or underwater) location of the reactor unit providing more protection from natural (*e.g.* seismic or tsunami according to the location) or man-made (*e.g.* aircraft impact) hazards.
- The modular design and small size lends itself to having multiple units on the same site.
- Lower requirement for access to cooling water – therefore suitable for remote regions and for specific applications such as mining or desalination.
- Ability to remove reactor module or in-situ decommissioning at the end of the lifetime.

In 2020 the IAEA published an update of its SMR book, Advances in Small Modular Reactor Technology Developments, with contributions from developers covering over 70 designs.

The IAEA has a programme assessing a conceptual multi-application small light water reactor (MASLWR) design with integral steam generators, focused on natural circulation of coolant, and in 2003 the US DOE published a report on this MASLWR conceptual design. Several of the integral PWR designs below have some similarities.

There are a number of small modular reactors coming forward requiring fuel enriched at the top end of what is defined as low-enriched uranium (LEU) – 20% U-235. The US Nuclear Infrastructure Council (NIC) has called for some of the downblending of military HEU to be only to about 19.75% U-235, so as to provide a small stockpile of fuel which would otherwise be very difficult to obtain (since civil enrichment plants normally cannot go above 5%). A reserve of 20 tonnes of high-assay low-enriched uranium (HALEU) has been suggested. The NIC said that the only supply of fuel for many advanced reactors under development would otherwise be foreign-enriched uranium. "Without a readily available domestic supply of higher enriched LEU in the USA, it will be extremely difficult to conduct research on advanced reactors, potentially driving American innovators overseas." In 2019 the DOE contracted with Centrus Energy to deploy a cascade of large centrifuges to produce HALEU fuel for advanced reactors. Urenco USA has announced its readiness to supply HALEU from a dedicated production line at its New Mexico plant.

US support for SMRs

In January 2012 the DOE called for applications from industry to support the development of one or two US light-water reactor designs, allocating $452 million over five years through the SMR Licensing Technical Support (LTS) programme. Four applications were made, from Westinghouse, Babcock & Wilcox, Holtec, and NuScale Power, the units ranging from 225 down to 45 MWe. The DOE announced its decision in November 2012 to support the B&W 180 MWe mPower design, to be developed with Bechtel and TVA. Through the five-year cost-share agreement, the DOE would invest up to half of the total project cost, with the project's industry partners at least matching this. The total would be negotiated between the DOE and B&W, and the DOE had paid $111 million by the end of 2014 before announcing that funds were cut off due to B&W shelving the project. However B&W is not required to repay any of the DOE money, and the project, capped at $15 million per year, is now under BWX Technologies. The company had spent more than $375 million on the mPower programme to February 2016.

In March 2012 the DOE signed agreements with three companies interested in constructing demonstration small reactors at its Savannah River site in South Carolina. The three companies and reactors are: Hyperion (now Gen4 Energy) with a 25 MWe fast reactor, Holtec with a 160 MWe PWR, and NuScale with its 45 MWe PWR (since increased to 60 MWe and then to 77 MWe – see below). The agreements concerned the provision of land but not finance. The DOE was in discussion with four further small reactor developers regarding similar arrangements, aiming to have in 10-15 years a suite of small reactors providing power for the DOE complex. (Over 1953-1991, Savannah River was where a number of production reactors for weapons plutonium and tritium were built and run.)

In March 2013 the DOE called for applications for second-round funding, and proposals were made by Westinghouse, Holtec, NuScale, General Atomics, and Hybrid Power Technologies, the last two being for EM2 and Hybrid SMR, not PWRs. Other (non-PWR) small reactor designs will have modest support through the Reactor Concepts RD&D programme. A late application "from left field" was from National Project Management Corporation (NPMC) which includes a cluster of regional partners in the state of New York, South Africa's PBMR company, and National Grid, the UK-based grid operator with 3.3 million customers in New York, Massachusetts and Rhode Island.*

* The project is for an HTR of 165 MWe, apparently the earlier direct-cycle version of the shelved PBMR, emphasising its 'deep burn' attributes in destroying actinides and achieving high burn-up at high temperatures. The PBMR design was a contender with Westinghouse backing for the US Next-Generation Nuclear Power (NGNP) project, which has stalled since about 2010.

In December 2013 the DOE announced that a further grant would be made to NuScale on a 50-50 cost-share basis, for up to $217 million over five years, to support design development and NRC certification and licensing of its initially 45 MWe small reactor design, subsequently increased to 60 MWe and then 77 MWe. In mid-2013 NuScale launched the Western Initiative for Nuclear (WIN) – a broad, multi-western state collaboration – to study the demonstration and deployment of multi-module NuScale SMR plants in the western USA. WIN includes Energy Northwest (ENW) in Washington and Utah Associated Municipal Power Systems (UAMPS). It is now called the Carbon-Free Power Project. A demonstration NuScale SMR built as part of Project WIN was projected to be operational by 2024, at the DOE's Idaho National Laboratory (INL), with UAMPS as the owner and ENW the operator. This would be followed by a full-scale (originally 12- but now six-module) plant there owned by UAMPS, run by Energy Northwest, and costing $5000/kW on an overnight basis, hence about $3.0 billion, with an expected levelized cost of electricity (LCOE) of $58/MWh from 2030.

In January 2014 Westinghouse announced that was suspending work on its small modular reactors in the light of inadequate prospects for multiple deployment. The company said that it could not justify the economics of its SMR without government subsidies, unless it could supply 30 to 50 of them. It was therefore delaying its plans, though small reactors remain on its agenda. In 2016 however, the company was much more positive about SMRs. See also UK Support subsection below. However, in March 2017 BWXT suspended work on the mPower design, after Bechtel withdrew from the project.

The Small Modular Reactor Research and Education Consortium (SmrREC) has been set up by Missouri University of Science and Technology to investigate the economics of deploying multiple SMRs in the country. SmrREC has constructed a comprehensive model of the business, manufacturing and supply chain needs for a new SMR-centric nuclear industry.

Early in 2016 developers and potential customers for SMRs set up the SMR Start consortium to advance the commercialization of SMR reactor designs. Members of the consortium include Bechtel, BWX Technologies, Dominion, Duke Energy, Energy Northwest, Fluor, GE Hitachi Nuclear Energy, Holtec, NuScale, Ontario Power, PSEG Nuclear, Southern Nuclear, Tennessee Valley Authority (TVA) and UAMPS. The organization will represent the companies in interactions with the US Nuclear Regulatory Commission (NRC), Congress and the executive branch on small reactor issues. US industry body the Nuclear Energy Institute (NEI) is assisting in the formation of the consortium, and is to work closely with the organization on policies and priorities relating to small reactor technology.

SMR Start has called for the DOE's LTS programme for SMRs to be extended to 2025 with an increase in funding. It pointed out: "Private companies and DOE have invested over $1 billion in the development of SMRs. However, more investment, through public-private partnerships is needed in order to assure that SMRs are a viable option in the mid-2020s. In addition to accomplishing the public benefit from SMR deployment, the federal government would receive a return on investment through taxes associated with investment, job creation and economic output over the lifetime of the SMR facilities that would otherwise not exist without the US government's investment."

In February 2016 TVA said it was still developing a site at Oak Ridge for a SMR and would apply for an early site permit (ESP, with no technology identified) for Clinch River in May with a view to building up to 800 MWe of capacity there. TVA has expanded discussions from B&W to include three other light-water SMR vendors. The DOE is supporting this ESP application financially from its SMR Licensing Technical Support Program, and in February 2016 DOE said it was committed to provide $36.3 million on cost-share basis to TVA.

In February 2021 TVA published a notice of intent to prepare a programmatic environmental impact statement on the potential effects of the construction, operation and decommissioning of an advanced nuclear reactor technology park at Clinch River. The park would contain one or more advanced nuclear reactors with a total electrical output of up to 800 MWe.

Another area of small reactor development is being promoted by the DOE's Advanced Research Projects Agency – Energy (ARPA-E) set up under a 2007 act. This focuses on high-potential, high-impact energy technologies that are too early for private-sector investment. ARPA-E is now beginning a new fission programme to examine microreactor technologies, below 10 MWe. This will solicit R&D project proposals for such reactors, which must have very high safety and security margins (including autonomous operations), be proliferation resistant, affordable, mobile, and modular. Targeted applications include remote sites, backup power, maritime shipping, military instillations, and space missions.

The DOE in 2015 established the Gateway for Accelerated Innovation in Nuclear (GAIN) initiative led by Idaho National Laboratory (INL) "to provide the new nuclear energy community with access to the technical, regulatory and financial support necessary to move new nuclear reactor designs toward commercialization. GAIN is based on feedback from the nuclear community and provides a single point of access to the broad range of capabilities – people, facilities, infrastructure, materials and data – across the Energy Department and its national laboratories." In January 2016 the DOE made grants of up to $40 million to X-energy for its Xe-100 pebble-bed HTR, and to Southern Company for the molten chloride fast reactor (MCFR) project being developed with TerraPower and Oak Ridge National Laboratory (ORNL).

In mid-2016 the DOE made GAIN grants of nuclear energy vouchers totalling $2 million including to Terrestrial Energy with Argonne National Laboratory, Transatomic Power with ORNL, and Oklo Inc with Argonne and INL for their respective reactor designs. A second round of GAIN voucher grants totalling $4.2 million was made in mid-2017, including to Terrestrial and Transatomic Power both with Argonne, Holtec's SMR Inventec for the SMR-160 at ORNL, Oklo Inc with Sandia and Argonne, and Elysium with INL and Argonne.

In April 2018, the DOE selected 13 projects to receive $60 million of cost-shared R&D funding for advance nuclear technologies, including the first awards under the US Industry Opportunities for Advance Nuclear Technology Development initiative.

In September 2018 the Nuclear Energy Innovation Capabilities Act and the Department of Energy Research and Innovation Act passed Congress. The first enables private and public institutions to carry out civilian research and development of advanced nuclear energy technologies. Specifically, the Act established the National Reactor Innovation Center to facilitate the siting of privately=funded advanced reactor prototypes at DOE sites through partnerships between the DOE and private industry. The second Act combines seven previously passed science bills to provide policy direction to the DOE on nuclear energy research and development.

In October 2018 the DOE announced that it was proposing to convert metallic high-assay low-enriched uranium (HALEU), with enrichment levels between 5% and 20% U-235, into fuel for research and development purposes. This would be at Idaho National Laboratory's Materials and Fuels Complex and/or the Idaho Nuclear Technology and Engineering Center, to support the development of new reactor technologies with higher efficiencies and longer core lifetimes.

The US Nuclear Regulatory Commission (NRC) has released a draft white paper on its strategy for reviewing licensing applications for advanced non-light water reactor technologies. The NRC said it expects to finalize the draft paper by November, with submission of the first non-LWR application expected by December 2019. By mid-2019 the NRC had been formally notified by six reactor designers of their intention to seek design approval. These included three MSRs, one HTR, one FNR, and the Westinghouse eVinci heatpipe reactor. In December 2019 the Canadian Nuclear Safety Commission (CNSC) and the US NRC selected Terrestrial Energy's Integral Molten Salt Reactor (IMSR) for the first joint technical review of an advanced, non-light water nuclear reactor.

In May 2020 the DOE launched the Advanced Reactor Demonstration Program (ARDP) offering funds, initially $160 million, on a cost-share basis for the construction of two advanced reactors that could be operational within seven years. The ARDP will concentrate resources on designs that are "affordable" to build and operate. The programme would also extend to risk reduction for future demonstrations, and include support under the Advanced Reactor Concepts 2020 pathway for innovative and diverse designs with the potential to be commercial in the mid-2030s. Testing and assessing advanced technologies would be carried out at the Idaho National Laboratory's National Reactor Innovation Center (NRIC). The NRIC started up in August 2019 as part of the DOE's Gateway for Accelerated Innovation in Nuclear (GAIN) initiative, which aims to accelerate the development and commercialization of advanced nuclear technologies. In October 2020 grants of $80 million each were made to TerraPower and X-energy to build demonstration plants that can be operational within seven years.

In December 2020 the DOE announced initial $30 million funding under the ARDP for five US-based teams developing affordable reactor technologies to be deployed over 10-14 years: Kairos Power for the Hermes Reduced-Scale Test Reactor, a scaled-down version of its fluoride salt-cooled high temperature reactor (KP-FHR); Westinghouse for the eVinci microreactor; BWXT Advanced Technologies for the BWXT Advanced Nuclear Reactor (BANR); Holtec for its SMR-160; and Southern Company for its Molten Chloride Reactor Experiment, a 300 kWt reactor project to provide data to inform the design of a demonstration molten chloride fast reactor (MCFR) using TerraPower's technology.

The DOE plans to build the Microreactor Applications Research Validation and Evaluation (MARVEL) reactor, a 100 kWt microreactor at Idaho. It is designed to perform research and development on various operational features of microreactors to improve their integration with end-user applications and is described in the Research Reactors information page.

In November 2021, among other advanced reactor projects, the DOE funded the second phase of a study on the potential for small reactors in Puerto Rico, at two suggested sites.

NuScale has announced that the DOE in 2022 would fund Ukraine's State Scientific and Technical Center for Nuclear and Radiation Safety to conduct an independent review of NuScale Power's safety analysis report for its SMR technology. The review will be accessible to any Ukrainian utility interested in deploying an SMR.

In August 2022 DOE Nuclear Energy University Program granted funds to CORE POWER and INL to research the economic and environmental benefits of floating advanced nuclear power generation.

In January 2023 NRC issued a final rule for the last stage in the design certification process, certifying NuScale Power's SMR, and allowing a utility to reference the design when applying for a combined licence to build and operate a nuclear power plant anywhere in the USA.

UK support for SMRs

The UK government in 2014 published a report on SMR concepts, feasibility and potential in the UK. It was produced by a consortium led by the National Nuclear Laboratory (NNL). Following this, a second phase of work is intended to provide the technical, financial and economic evidence base required to support a policy decision on SMRs. If a future decision was to proceed with UK development and deployment of SMRs, then further work on the policy and commercial approach to delivering them would need to be undertaken, which could lead to a technology selection process for UK generic design assessment (GDA).

In March 2016 the UK Department of Energy & Climate Change (DECC) called for expressions of interest in a competition to identify the best value SMR for the UK. This relates to a government announcement in November 2015 that it would invest at least £250 million over five years in nuclear R&D including SMRs. DECC said the objective of the initial phase was "to gauge market interest among technology developers, utilities, potential investors and funders in developing, commercializing and financing SMRs in the UK." It said the initial stage would be a "structured dialogue" between the government and participants, using a published set of criteria, including that the SMR design must "be designed for manufacture and assembly, and … able to achieve in-factory production of modular components or systems amounting to a minimum of 40% of the total plant cost."

In December 2017, the Department for Business, Energy & Industrial Strategy (BEIS), DECC's successor department, announced that the SMR competition had been closed. Instead, a new two-phase advanced modular reactor competition was launched, designed to incorporate a wider range of reactor types. Total funding for the Advanced Modular Reactor (AMR) Feasibility and Development (F&D) project is up to £44 million, and 20 bids had been received by the initial deadline of 7 February 2018. In September 2018 it was announced that the following eight organisations were awarded contracts up to £300,000 to produce feasibility studies for the first phase of the AMR F&D project: Advanced Reactor Concepts (ARC-100); DBD (representing China's Institute of Nuclear and New Energy Technology's HTR-PM); LeadCold (SEALER-UK); Moltex Energy (Stable Salt Reactor); Tokamak Energy (compact spherical modular fusion reactor); U-Battery Developments (U-Battery); Ultra Safe Nuclear (Micro-Modular Reactor); and Westinghouse (Westinghouse LFR).

In July 2020, under its AMR programme, BEIS awarded £10 million to each of: Westinghouse, for its 450 MWe LFR; U-Battery consortium for its 4 MWe HTR; and Tokamak Energy for its compact fusion reactor project. A further £5 million will be for British companies and start-ups to develop new ways of manufacturing advanced nuclear parts for modular reactor projects both at home and abroad. Another £5 million is to strengthen the country's nuclear regulatory regime as it engages with advanced nuclear technologies such as these.

In March 2019 BEIS released a 2016 report on microreactors that defined them as having a capacity up to 100 MWt/30 MWe, and projecting a global market for around 570 units of an average 5 MWe by 2030, total 2850 MWe. It notes that they are generally not water-moderated or water cooled, but "use a compact reactor and heat exchange arrangement, frequently integrated in a single reactor vessel." Most are HTRs.

In 2015 Westinghouse had presented a proposal for a "shared design and development model" under which the company would contribute its SMR conceptual design and then partner with UK government and industry to complete, license and deploy it. The partnership would be structured as a UK-based enterprise jointly owned by Westinghouse, the UK government and UK industry. In October 2016 the company said it would work with UK shipbuilder Cammell Laird as well as the UK's Nuclear Advanced Manufacturing Research Centre (NAMRC) on a study to explore potential design efficiencies to reduce the lead times of its SMR.

NuScale has said that it aims to deploy its SMR technology in the UK with UK partners, so that the first of its units could be in operation by the mid-2020s. In September 2017 the company released its five-point UK SMR action plan. Rolls-Royce submitted a detailed design to the government for a 220 MWe SMR unit.

In November 2021 the UK government announced that it would contribute £210 million in grant funding to Rolls-Royce SMR to match private investment in this venture. Rolls-Royce Group, BNF Resources UK and Exelon Generation will invest £195 million over about three years in it. Rolls-Royce said the SMR business, which will continue to seek further investment, will now "proceed rapidly with a range of parallel delivery activities, including entry to the UK generic design assessment (GDA) process and identifying sites for the factories which will manufacture the modules that enable onsite assembly of the power plants." The reactor is designed for hydrogen and synthetic fuel manufacturing as well as electricity generation. The Rolls-Royce SMR consortium, involving many of the major UK engineering firms, aims to build 16 reactors, each a pressurized water type of 470 MWe.

In November 2022, Rolls-Royce announced that it has identified four priority locations to build SMR-based power stations in the UK, including Trawsfynydd, Wylfa, and Oldbury. The locations are all on land owned by the UK Nuclear Decommissioning Authority (NDA). Before NDA commits to the SMR development, approval must first be granted by the Department of Business, Energy, and Industrial Strategy.

Canadian support for SMRs

A June 2016 report for the Ontario Ministry of Energy focused on nine designs under 25 MWe for off-grid remote sites. All had a medium level of technology readiness and were expected to be competitive against diesel. Two designs were integral PWRs of 6.4 and 9 MWe, three were HTRs of 5, 8 and 16 MWe, two were sodium-cooled fast reactors (SFRs) of 1.5/2.8 and 10 MWe, one was a lead-cooled fast reactor (LFR) of 3-10 MWe, and one was an MSR of 32.5 MWe. Four were under 5 MWe (an SFR, LFR, and two HTRs). Ontario distinguishes 'grid scale' SMRs above 25 MWe from these (very) small-scale reactors.

The Canadian Nuclear Safety Commission (CNSC) has been conducting pre-licensing vendor design reviews – an optional service to assess a nuclear power plant design based on a vendor's reactor technology – for ten* small reactors with capacities in the range of 3-300 MWe. Two further agreements for design review are being negotiated for StarCore's HTR and Westinghouse's eVinci. In May 2021 it commenced a formal licence review of the 15 MWt MMR-5 for Global First Power (a joint venture between Ultra Safe Nuclear Corporation and Ontario Power Generation).

* Terrestrial Energy’s IMSR; USNC’s MMR-5 and MMR-10; LeadCold Nuclear’s SEALER; ARC Nuclear’s ARC-100; Moltex’s Stable Salt Reactor; SMR’s SMR-160; NuScale’s Power Module; U-Battery's U-Battery, GE Hitachi's BWRX-300; X-energy's Xe-100.

In June 2017 Canadian Nuclear Laboratories (CNL) invited expressions of interest in SMRs. This resulted in many responses, including 19 for siting a demonstration or prototype reactor at a CNL-managed site. CNL aims to have a new SMR at its Chalk River site by 2026. Global First Power with its partners Ontario Power Generation and Ultra-Safe Nuclear Corporation was the first to get to the third stage of CNL’s siting evaluation, with its MMR, a 5 MWe HTR. In February 2019 CNL announced that StarCore Nuclear and Terrestrial Energy had qualified to enter the due diligence (second) stage of its siting evaluation for their 14 MWe HTR and 195 MWe IMSR respectively.

In November 2019 CNL announced that Kairos Power, Moltex Canada, Terrestrial Energy and Ultra Safe Nuclear Corporation (USNC) had been selected as the first recipients of support under its Canadian Nuclear Research Initiative (CNRI). This is designed to accelerate SMR deployment by enabling research and development on particular projects and connecting global vendors of SMR technology with the facilities and expertise within Canada's national nuclear laboratories. Recipients are expected to match the value contributed by CNL either in monetary or in-kind contributions.

In November 2018 the Canadian government released its SMR Roadmap, a 10-month nationwide study of SMR technology. The report concludes that Generation IV SMR development is a response to market forces for "smaller, simpler and cheaper" nuclear energy, and the large global market for this technology will be "driven not just by climate change and clean energy policies, but also by the imperatives of energy security and access." In October 2020 the Minister for Innovation, Science & Industry announced a C$20 million investment in Terrestrial Energy to accelerate development of its Integral Molten Salt Reactor (IMSR), the first grant from Canada’s Strategic Innovation Fund.

In December 2019 Saskatchewan and New Brunswick agreed to work with Ontario in promoting SMRs to "unlock economic potential across Canada, including rural and remote regions" in line with the national SMR Roadmap. In August 2020 Alberta joined in, flagging the potential for SMRs to be used for the province's northern oil sands industry. The agreement is to also address key issues for SMR deployment including technological readiness, regulatory frameworks, economics and financing, nuclear waste management and public and indigenous engagement. In 2021 Alberta’s largest oil sands producers formed an alliance to consider ways to achieve net zero greenhouse gas emissions by 2050, with SMRs being part of the means.

In October 2020 Ontario Power Generation (OPG) announced that it would take forward engineering and design work with three developers of grid-scale SMRs – GE Hitachi (GEH), Terrestrial Energy and X-energy – to support remote area energy needs. The focus is on GEH’s 300 MWe BWRX-300, Terrestrial’s 192 MWe Integral Molten Salt Reactor, and X-energy’s 80 MWe Xe-100 high-temperature SMRs. All three are in phase 2 of the CNSC’s vendor design review process. GEH is setting up a Canadian supply chain for its BWRX-300.

In November 2020 New Brunswick Power and Moltex Energy were joined by ARC Canada in setting up an SMR vendor cluster at Point Lepreau, and in March 2021 the Canadian government announced C$56 million support for this, mostly for the Moltex Stable Salt Reactor – Wasteburner (SSR-W) project.

Chinese support for SMRs

The most advanced small modular reactor project is in China, where Chinergy is starting to build the 210 MWe HTR-PM, which consists of twin 250 MWt high-temperature gas-cooled reactors (HTRs) which build on the experience of several innovative reactors in the 1960s to 1980s.

CNNC New Energy Corporation, a joint venture of CNNC (51%) and China Guodian Corp, is promoting the ACP100 reactor. A preliminary safety analysis report for a single unit demonstration plant at Changjiang was approved in April 2020.

However, China is also developing small district heating reactors of 100 to 200 MWt capacity which may have a strong potential evaluated at around 400 units. The heat market is very large in northern China, now almost exclusively served by coal, causing serious pollution, particularly by dust, particulates, sulfur, and nitrogen oxides.

Overall SMR research and development in China is very active, with vigorous competition among companies encouraging innovation.

Other countries

Urenco has called for European development of very small – 4 MWe – 'plug and play' inherently-safe reactors based on graphite-moderated HTR concepts. It is seeking government support for a prototype "U-Battery" which would run for 5-10 years before requiring refuelling or servicing.

Already operating in a remote corner of Siberia are four small units at the Bilibino co-generation plant. These four 62 MWt (thermal) units are an unusual graphite-moderated boiling water design with water/steam channels through the moderator. They produce steam for district heating and 11 MWe (net) electricity each, remote from any grid. They are the world's smallest commercial power reactors and have performed well since 1976, much more cheaply than fossil fuel alternatives in the severe climate of this Arctic region, but are due to be retired by 2023.

Looking ahead, and apart from its barge-mounted ones, Rosatom is not positive about small reactors generally.

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

Another significant line of development is in very small fast reactors of under 50 MWe. Some are conceived for areas away from transmission grids and with small loads; others are designed to operate in clusters in competition with large units.

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

In December 2019 CEZ in the Czech Republic said it was focusing on 11 SMR designs including these seven: Rosatom's RITM-200, GE Hitachi Nuclear Energy's BWRX-300, NuScale Power's SMR, China National Nuclear Corporation's ACP100, Argentina's CAREM, the South Korean SMART, and Holtec International's SMR-160.

Small reactors operating

Name	Capacity	Type	Developer
CNP-300	300 MWe	PWR	SNERDI/CNNC, Pakistan & China
PHWR-220	220 MWe	PHWR	NPCIL, India
EGP-6	11 MWe	LWGR	at Bilibino, Siberia (cogen, soon to retire)
KLT-40S	35 MWe	PWR	OKBM, Russia
RITM-200	50 MWe	Integral PWR, civil marine	OKBM, Russia

Small reactor designs under construction

Name	Capacity	Type	Developer
CAREM25	27 MWe	Integral PWR	CNEA & INVAP, Argentina
HTR-PM	210 MWe	Twin HTR	INET, CNEC & Huaneng, China
ACP100/Linglong One	125 MWe	Integral PWR	CNNC, China
BREST	300 MWe	Lead FNR	RDIPE, Russia

Small reactors for near-term deployment – development well advanced

Name	Capacity	Type	Developer
VBER-300	300 MWe	PWR	OKBM, Russia
NuScale Power Module	77 MWe	Integral PWR	NuScale Power + Fluor, USA
SMR-160	160 MWe	PWR	Holtec, USA + SNC-Lavalin, Canada
SMART	100 MWe	Integral PWR	KAERI, South Korea
BWRX-300	300 MWe	BWR	GE Hitachi, USA
PRISM	311 MWe	Sodium FNR	GE Hitachi, USA

Name	Capacity	Type	Developer
Natrium	345 MWe	Sodium FNR	TerraPower + GE Hitachi, USA
ARC-100	100 MWe	Sodium FNR	ARC with GE Hitachi, USA
Integral MSR	192 MWe	MSR	Terrestrial Energy, Canada
Seaborg CMSR	100 MWe	MSR	Seaborg, Denmark
Hermes prototype	35 MWt	MSR-Triso	Kairos, USA
RITM-200M	50 MWe	Integral PWR	OKBM, Russia
RITM-200N	55 MWe	Integral PWR	OKBM, Russia
BANDI-60S	60 MWe	PWR	Kepco, South Korea
Xe-100	80 MWe	HTR	X-energy, USA
ACPR50S	60 MWe	PWR	CGN, China
Moltex SSR-W	300 MWe	MSR	Moltex, UK

Small reactor designs at earlier stages (or shelved)

Name	Capacity	Type	Developer
EM2	240 MWe	HTR, FNR	General Atomics (USA)
FMR	50 MWe	HTR, FNR	General Atomics + Framatome
VK-300	300 MWe	BWR	NIKIET, Russia
AHWR-300 LEU	300 MWe	PHWR	BARC, India
CAP200 LandStar-V	220 MWe	PWR	SNERDI/SPIC, China
SNP350	350 MWe	PWR	SNERDI, China
ACPR100	140 MWe	Integral PWR	CGN, China
IMR	350 MWe	Integral PWR	Mitsubishi Heavy Ind, Japan*
Westinghouse SMR	225 MWe	Integral PWR	Westinghouse, USA*
mPower	195 MWe	Integral PWR	BWXT, USA*
UK SMR	470 MWe	PWR	Rolls-Royce SMR, UK
PBMR	165 MWe	HTR	PBMR, South Africa*
HTMR-100	35 MWe	HTR	HTMR Ltd, South Africa
MCFR	large?	MSR/FNR	Southern Co, TerraPower, USA
SVBR-100	100 MWe	Lead-Bi FNR	AKME-Engineering, Russia*
Westinghouse LFR	300 MWe	Lead FNR	Westinghouse, USA
TMSR-SF	100 MWt	MSR	SINAP, China
PB-FHR	100 MWe	MSR	UC Berkeley, USA
Moltex SSR-U	150 MWe	MSR/FNR	Moltex, UK
Thorcon TMSR	250 MWe	MSR	Martingale, USA
Leadir-PS100	36 MWe	Lead-cooled	Northern Nuclear, Canada

Very small reactor designs being developed (up to 25 MWe)

Name	Capacity	Type	Developer
U-battery	4 MWe	HTR	Urenco-led consortium, UK
Starcore	10-20 MWe	HTR	Starcore, Quebec
MMR-5/-10	5 or 10 MWe	HTR	UltraSafe Nuclear, USA
Holos Quad	3-13 MWe	HTR	HolosGen, USA
Gen4 module	25 MWe	Lead-bismuth FNR	Gen4 (Hyperion), USA

Name	Capacity	Type	Developer
Xe-Mobile	1-5 MWe	HTR	X-energy, USA
BANR	50 MWt	HTR	BWXT, USA
Sealer	3-10 MWe	Lead FNR	LeadCold, Sweden
eVinci	0.2-5 MWe	Heatpipe FNR	Westinghouse, USA
Aurora	1.5 MWe	Heatpipe FNR	Oklo, USA
NuScale micro	1-10 MWe	Heatpipe	NuScale, USA

See also IAEA Advances in Small Modular Reactor Technology Developments, A Supplement to: IAEA Advanced Reactors Information system (ARIS), 2020 Edition.

** Well-advanced designs understood to be on hold or abandoned.*

Military developments of small power reactors from 1950s

US experience and plans

About five decades ago the US Army built eight reactors, five of them portable or mobile. PM1 successfully powered a remote air/missile defence radar station on a mountain top near Sundance, Wyoming for six years to 1968, providing 1 MWe. At Camp Century in northern Greenland the 10 MWt, 1.56 MWe plus 1.05 GJ/hr PM-2A was assembled from prefabricated components, and ran from 1960-64 on high-enriched uranium fuel. Another was the 9 MWt, 1.5 MWe (net) PM-3A reactor which operated at McMurdo Sound in Antarctica from 1962-72, generating a total of 78 million kWh and providing heat. It used high-enriched uranium fuel and was refuelled once, in 1970. MH-1A was the first floating nuclear power plant operating in the Panama Canal Zone from 1968-77 on a converted Liberty ship. It had a 45 MWt/10 MWe (net) single-loop PWR which used low-enriched uranium (4-7%). It used 541 kg of U-235 over ten years and provided power for nine years at 54% capacity factor.

ML-1 was a smaller and more innovative 0.3 MWe mobile power plant with a water-moderated HTR using pressurized nitrogen at 650°C to drive a Brayton closed cycle gas turbine. It used HEU in a cluster of 19 pins, the core being 56 cm high and 56 cm diameter. It was tested over 1962-66 in Idaho. It was about the size of a standard shipping container and was truck-mobile and air-transportable, with 12-hour set-up. The control unit was separate, to be located 150 m away.

All these were outcomes of the Army Nuclear Power Program (ANPP) for small reactor development – 0.1 to 40 MWe – which ran from 1954-77. ANPP became the Army Reactor Office (ARO) in 1992. More recently (2010) the DEER (Deployable Electric Energy Reactor) was being commercialized by Radix Power & Energy for forward military bases or remote mining sites. See later subsection.

A 2018 report from the US Army analysed the potential benefits and challenges of mobile nuclear power plants (MNPPs) with very small modular reactor (vSMR) technology. This followed a 2016 report on Energy Systems for Forward/Remote Operating Bases. The purpose is to reduce supply vulnerabilities and operating costs while providing a sustainable option for reducing petroleum demand and consequent vulnerability. MNPPs would be portable by truck or large aircraft and if abroad, returned to the USA for refuelling after 10-20 years. They would load-follow and run on low-enriched uranium (<20%), probably as TRISO (tristructural-isotropic) fuel in high-temperature gas-cooled reactors (HTRs).

In January 2019 the Department of Defense (DOD) Strategic Capabilities Office solicited proposals for a 'small mobile reactor' design which could address electrical power needs in rapid response scenarios – Project Pele. These would make domestic infrastructure resilient to an electrical grid attack and change the logistics of forward operating bases, both by making more energy available and by simplifying fuel logistics needed to support existing, mostly diesel-powered, generators. They would also enable a more rapid response during humanitarian assistance and disaster relief operations. "Small mobile nuclear reactors have the potential to be an across-the-board strategic game changer for the DOD by saving lives, saving money, and giving soldiers in the field a prime power source with increased flexibility and functionality." The reactors need to be designed to be operated by a crew of six, with one fully qualified engineer and a single operator on duty at all times.

Each reactor should be an HTR with high-assay low-enriched uranium (HALEU) TRISO fuel and produce a threshold power of 1-10 MWe for at least three years without refuelling. It must weigh less than 40 tonnes and be sized for transportability by truck, ship, and C-17 aircraft. Designs must be "inherently safe", ensuring that a meltdown is "physically impossible" in various complete failure scenarios such as loss of power or cooling, and must use ambient air as their ultimate heat sink, as well as being capable of passive cooling. The reactor must be capable of being installed to the point of "adding heat" within 72 hours and of completing a planned shutdown, cool down, disconnect and removal of transport in under seven days. The DOD announced its preparation of an environmental impact statement for the reactor in March 2020, and awarded $12-14 million contracts to three companies for initial design work. Then BWXT Advanced Technologies and X-energy were selected in March 2021 to develop a final engineering design by March 2022. Westinghouse has dropped out, and one of the two companies may be commissioned in 2022 to build a prototype reactor.

The DOD in March 2021 said Project Pele is on track for full power testing of a mobile reactor in 2023, with outdoor mobile testing of a prototype microreactor built at Idaho National Laboratory or Oak Ridge National Laboratory in 2024. The programme is also intended to spur commercial development of HTRs. In September 2021 the DOD issued a draft environmental impact statement for the construction and demonstration operation of a prototype mobile microreactor.

In October the US Air Force announced that its first microreactor would be at Eielson air force base in Alaska, near Fairbanks, to be operational in 2027. This does not appear to be part of Project Pele. The base has its own 15 MWe coal-fired power station already, with a railway to supply it with fuel.

Russian experience

The Joint Institute for Power Engineering and Nuclear Research (Sosny) in Belarus built two Pamir-630D truck-mounted small air-cooled nuclear reactors in 1976, during the Soviet era. The entire plant required several trucks. This was a 5 MWt/0.6 MWe HTR reactor using 45% enriched fuel with zirconium hydride moderator and driving a gas turbine with dinitrogen tetroxide through the Brayton cycle. After some operational experience the Pamir project was scrapped in 1985-86. It had been preceded by the 1.5 MWe TES-3, a PWR mounted on four heavy tank chassis, each self-propelled, with the modules (reactor, steam generator, turbine, control) coupled onsite. The prototype started up in 1961 at Obninsk, operated to 1965, and was abandoned in 1969.

Since 2010 Sosny has been involved with Luch Scientific Production Association (SRI SIA Luch) and Russia's N.A. Dollezhal Research and Development Institute of Power Engineering (NIKIET or RDIPE) to design a small transportable nuclear reactor. The new design will be an HTR concept similar to Pamir but about 2.5 MWe.

A small Russian HTR which was being developed by NIKIET is the Modular Transportable Small Power Nuclear Reactor (MTSPNR) for heat and electricity supply of remote regions. It is described as a single circuit air-cooled HTR with closed cycle gas turbine. It uses 20% enriched fuel and is designed to run for 25 years without refuelling. A twin unit plant delivers 2 MWe and/or 8 GJ/h. It is also known as GREM. No recent information is available, but an antecedent is the Pamir, from Belarus. More recently NIKIET has described the ATGOR – a transportable HTR with up to six parallel commercial gas-turbine engines with two independent heat sources (a nuclear reactor and a start-up diesel fuelled combustor).

Another NIKIET project is the 6 MWt, 1 MWe Vityaz modular integral light water reactor with two turbine generators, which is transportable as four modules of up to 60 tonnes.

In 2015 it was reported that the Russian defence ministry had commissioned the development of small mobile nuclear power plants for military installations in the Arctic. A pilot project being undertaken by Innovation Projects Engineering Company (IPEC) is a mobile low-power nuclear unit to be mounted on a large truck, tracked vehicle or a sledged platform. Production models will need to be capable of being transported by military cargo jets and heavy cargo helicopters, such as the Mil Mi-26. They need to be fully autonomous and designed for years-long operation without refuelling, with a small number of personnel, and remote control centre. It is assumed but not confirmed that these reactors will be the MTSPNR.

Temperatures of small reactors

Many small reactors are designed for industrial heat applications as well as power generation. So, while light water reactors are constrained by pressure limitations and thus operate in the 300-400°C range, others are higher temperature. Liquid metal fast reactors are in the 400-600°C range, molten salt reactors are around 600-700°C, and high-temperature reactors are 600-900°C.

Light water reactors

These are moderated and cooled by ordinary water and have the lowest technological risk, being similar to most operating power and naval reactors today. They mostly use fuel enriched to less than 5% U-235 with no more than a six-year refuelling interval, and regulatory hurdles are likely least of any small reactors.

US experience of small light water reactors (LWRs) has been of small military power plants, mostly PWRs – see above.

Some successful small reactors from the main national programme commenced in the 1950s. One was the Big Rock Point BWR of 67 MWe which operated for 35 years to 1997.

The US Nuclear Regulatory Commission is starting to focus on small light-water reactors using conventional fuel, such as B&W, Westinghouse, NuScale, and Holtec designs including integral types (B&W, Westinghouse, NuScale). Beyond these in time and scope, “the NRC intends to take full advantage of the experience and expertise” of other nations which have moved forward with non light-water designs, and it envisages “having a key role in future international regulatory initiatives.”

Of the following designs, the KLT, VBER and Holtec SMR have conventional pressure vessels plus external steam generators (PV/loop design). The others mostly have the steam supply system inside the reactor pressure vessel ('integral' PWR design). All have enhanced safety features relative to current LWRs. All require conventional cooling of the steam condenser.

In the USA major engineering and construction companies have taken active shares in two projects: Fluor in NuScale, and Bechtel in B&W mPower.

Three new concepts are alternatives to conventional land-based nuclear power plants. Russia's floating nuclear power plant (FNPP) with a pair of PWRs derived from icebreakers is well on the way to commissioning, with the KLT-40S reactors described below and in the Nuclear Power in Russia information page. The next generation is expected to use RITM-200M reactors. China has a similar project for its ACP100 SMR as a FNPP, whilst MIT is developing a floating plant moored offshore with a reactor of about 200 MWe in the bottom part of a cylindrical platform. France's submerged Flexblue power plant, using a 50-250 MWe reactor, was an early concept but is now cancelled.

KLT-40S

Russia's KLT-40S from OKBM Afrikantov is derived from the KLT-40 reactor well proven in icebreakers and now – with low-enriched fuel – on a barge, for remote area power supply. Here a 150 MWt unit produces 35 MWe (gross) as well as up to 35 MW of heat for desalination or district heating (or 38.5 MWe gross if power only). Burn-up is 45 GWd/t. Units are designed to run 3-4 years between refuelling with on-board refuelling capability and used fuel storage. All fuel assemblies are replaced in each such refuelling. At the end of a 12-year operating cycle the whole plant is taken to a central facility for overhaul and storage of used fuel. Operating plant lifetime is 40 years. Two units are mounted on a 21,500 tonne barge.

Although the reactor core is normally cooled by forced circulation (four-loop), the design relies on convection for emergency cooling. Fuel is uranium aluminium silicide with enrichment levels of 18.6%, giving three-year refuelling intervals. A variant of this is the KLT-20, specifically designed for floating nuclear plants. It is a two-loop version with the same enrichment but with a ten-year refuelling interval.

The first floating nuclear power plant, the *Akademik Lomonosov*, commenced construction in 2007, and was grid connected at Pevek in December 2019. (See also *Floating nuclear power plants* section in the information page on Nuclear Power in Russia.)

RITM-200M, RITM-200N

The RITM series is Russia's 'flagship' SMR design. The compact RITM-200M will replace the KLT reactors to serve in floating nuclear power plants, or optimized floating power units (OFPUs) as they are now called by OKBM. It is derived from the OKBM Afrikantov's RITM-200 reactor units in the LK-60 icebreakers and is an integral 175 MWt/50 MWe PWR with 12 steam generator cassettes inside the pressure vessel and four coolant loops with external main circulation pumps. It has inherent safety features, using low-enriched (<20%) fuel in 241 fuel assemblies (compared with 199 in the icebreaker version). OFPUs will be returned to base for servicing every 10 or 12 years and no onboard used fuel storage is required. Operational lifetime is 60 years. Each reactor can supply 730 GJ/h thermal power. Twin reactor units in containment have a mass of 2600 tonnes and occupy 6.8 m × 14.6 m × 16.0 m high, requiring only a 12,000 tonne barge – much smaller than the KLT-40S units. A major challenge is the reliability of steam generators and associated equipment which are much less accessible when inside the reactor pressure vessel.

Rosatom is planning three OFPUs each with twin RITM-200M reactors at Cape Nagloynyn to supply 330 MWe to the Baimskaya copper mining project south of Bilibino and Pevek.

Onshore installation of the similar RITM-200N is also envisaged, with one or more modules of 190 MWt/55 MWe, fuel enriched to almost 20% and 5-6 year fuel cycle. Reactor containment dimensions are 6 m × 6 m × 15.5 m. The first plant is to be in Ust-Kuyga in Yakutia. Rostechnadzor licensed this in August 2021, with construction to begin in 2024 and operation expected in 2028. This will be a reference plant for export sales.

The RITM-200B is a 209 MWt version and the RITM-400 is a 315 MWt version, both for icebreaker use.

CNP-300

This is based on the early Qinshan 1 reactor in China as a two-loop PWR, with four operating in Pakistan. It is 1000 MWt, 325 MWe with a design operating lifetime of 40 years. Fuel enrichment is 2.4-3.0%, with refuelling at 12-month intervals. It was designed by Shanghai Nuclear Energy Research & Design Institute (SNERDI).

SNP350

The SNP350 is SNERDI's development of the CNP-300, upgraded in many respects to meet latest performance, economy, and safety requirements. It is 1035 MWt, 350 MWe gross, with design operating lifetime of 60 years and digital I&C systems.

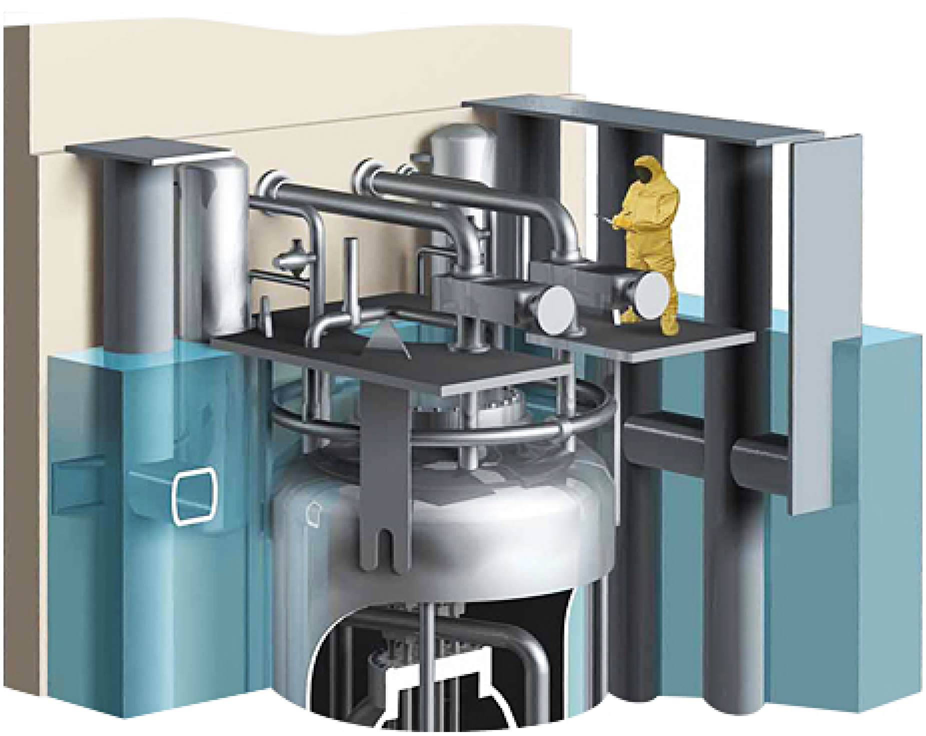
NuScale

The NuScale Power Module is a 250 MWt, 77 MWe gross integral PWR with natural circulation.* In December 2013 the US Department of Energy (DOE) announced that it would support accelerated development of the design for early deployment on a 50-50 cost share basis. An agreement for $217 million over five years was signed in May 2014 by NuScale Power. In September 2017, following acceptance of the company's design certification application (DCA) by the US Nuclear Regulatory Commission (NRC) earlier in the year, NuScale applied for the second part of its loan guarantee with the US DOE.

* In November 2020, it was announced that "further value engineering efforts" had resulted in the capacity of the NuScale Power Module being 25% higher than its previous value of 200 MWt, 60 MWe gross.

It will be factory-built with a three-metre diameter pressure vessel and convection cooling, with the only moving parts being the control rod drives. It uses standard PWR fuel enriched to 4.95% in normal PWR fuel assemblies (but which are only 2 m long), with 24-month refuelling cycle. Installed in a water-filled pool below ground level, the 4.6 m diameter, 23 m high cylindrical containment vessel module weighs 640 tonnes and contains the reactor with steam generator above it. A standard power plant would have 12 modules together giving about 924 MWe, though four-module and six-module plants are now envisaged also. The multi-unit plants are called VOYGR. An overhead crane would hoist each module from its pool to a separate part of the plant for refuelling. Design operational lifetime is 60 years. It has full passive cooling in operation and after shutdown for an indefinite period, without even DC battery requirement. The NRC concluded in January 2018 that NuScale's design eliminated the need for class 1E backup power – a current requirement for all US nuclear plants. It claims good load-following capability, in line with EPRI requirements and also black start capability.

The UK's National Nuclear Laboratory (NNL) has confirmed that the reactor can run on MOX fuel. It also said that a VOYGR-12 plant with full MOX cores could consume 100 tonnes of reactor-grade plutonium in about 40 years, generating 200 TWh from it. This would be in line with Areva's proposal for using the UK plutonium stockpile, especially since Areva is already contracted to make fuel for the NuScale reactor.



NuScale Power Module (NuScale)

The company had estimated in 2010 that overnight capital cost for a 12-module, 540 MWe plant would be about $4000 per kilowatt, this in 2014 had risen to $5078/kWe net, with the levelized cost of electricity (LCOE) expected to be $100/MWh for first unit (or $90 for '*n*th-of-a-kind'). In June 2018, the company announced that its reactor can generate 20% more power than originally planned. Subject to NRC approval, this would lower the overnight capital cost to about $4200 per kilowatt, and lower the LCOE by 18%. With a further power increase late in 2020 the company quoted a capital cost of $2850/kWe (for a 12-module 924 MWe plant).

The NuScale Power company was spun out of Oregon State University in 2007, though the original development was funded by the US Department of Energy. After NuScale experienced problems in funding its development, Fluor Corporation paid over $30 million for 55% of NuScale in October 2011. In May 2022 NuScale Power announced that it had merged with Spring Valley Acquisition Corp. The combined company, NuScale Power Corporation, is listed on the NYSE. Fluor continues to hold a majority interest in the company, and provide it with engineering services, project management, and administration and supply chain support.

In April 2012 ARES Corporation agreed to assist in design and licensing. March 2014 Enercon Services became a partner to assist with design certification and licence applications. In October 2015 Ultra Electronics agreed to contribute technical expertise. In July 2019 Doosan Heavy Industries brought its pressure vessel manufacturing ability to the project and followed this with $104 million equity. Also in July 2019 Sargent & Lundy agreed to support the plant design. In April 2021 Japan’s JGC Holdings agreed to invest $40 million and, as EPC contractor, to partner with Fluor in deployment of NuScale SMRs. In May 2021 Japan’s IHI invested $20 million cash and became a strategic partner. In June 2021 GS Energy North America joined them, as did Samsung in July. All these contributed equity to NuScale, though leaving Fluor as majority and lead strategic investor.

NuScale lodged an application for US design certification in January 2017, and in July 2017 the NRC confirmed that its highly integrated protection system (HIPS) architecture was approved. NuScale has been engaged with the NRC since 2008, having spent some $130 million on licensing to November 2013. In September 2020 the NRC issued a standard design approval for the earlier 50 MWe version.* NuScale said it would apply in 2022 for the same approval for the 60 MWe version, although later, in November 2020, the company announced that each module would now be 77 MWe. It is the first SMR to receive NRC design approval. In October 2022 the NRC said it agreed with NuScale’s methodology for calculating the emergency planning zone (EPZ) acceptable for use with NuScale’s design.

* The standard design approval (SDA) allows the NuScale standard design to be referenced in an application for a construction permit or operating licence, or an application for a combined construction and operating licence (COL) under NRC regulations. Site-specific licensing procedures must also be completed before any construction can begin.

In September 2018 NuScale selected BWX Technologies as the first manufacturer of its SMR after an 18-month selection process. The demonstration unit in Idaho will have dry cooling for the condenser circuit, with a 90% water saving while sacrificing about 5% of its power output to drive the cooling. In mid-2021 Doosan said it was preparing to start the forging fabrication for UAMPS reactor modules in 2022 and Samsung said that NuScale, Fluor and Samsung C&T Corporation would work together to deliver NuScale plants globally.

In December 2019 NuScale submitted its 60 MWe (now 77 MWe) SMR design to the Canadian Nuclear Safety Commission (CNSC) for pre-licensing vendor design review. Phase 2 of this commenced in January 2020.

Earlier in March 2012 the DOE signed an agreement with NuScale regarding constructing a demonstration unit at its Savannah River Site in South Carolina.

In mid-2013 NuScale launched the Western Initiative for Nuclear (WIN) – a broad, multi-western state collaboration* – to study the demonstration and deployment of a multi-module NuScale SMR plant in western USA. This became the **Carbon-Free Power Project** led by Utah Associated Municipal Power Systems (UAMPS) at the DOE’s Idaho National Laboratory (INL). With the unit power to increase to 77 MWe, the overnight capital cost of a six-module plant would be about $3 billion, hence $6500/kW. UAMPS has 27 public utilities participating in the project. UAMPS is targeting $58/MWh generation cost (LCOE) for a six-module plant. The first unit is expected to be online in 2029.

WIN includes Energy Northwest (ENW) in Washington and Utah Associated Municipal Power Systems (UAMPS). A demonstration NuScale SMR built as part of Project WIN is projected to be operational in 2029, at the DOE’s Idaho National Laboratory (INL), with UAMPS as the owner and ENW the operator. This would be followed by a full-scale (originally planned as 12- but now six-module) plant owned by UAMPS and run by Energy Northwest. With the unit power to increase to 77 MWe, the cost of a 12-module plant would be about $2850/kW on an overnight basis. Energy Northwest comprises 27 public utilities, and had examined small reactor possibilities before choosing NuScale and becoming part of the **UAMPS Carbon-Free Power Project**. UAMPS is targeting $55/MWh generation cost (LCOE).

* Washington, Oregon, Idaho, Wyoming, Utah and Arizona.

In Poland, NuScale is exploring with Unimot and KGHM possibilities for its reactors to replace coal-fired power plants.

NuScale is investigating cogeneration options including desalination (with Aquatech), oil recovery from tar sands and refinery power (with Fluor), hydrogen production by high-temperature steam electrolysis (with INL) and flexible back-up for a wind farm (with UAMPS and Energy Northwest). Doosan is cooperating on hydrogen production and desalination.

NuScale and Prodigy Clean Energy are developing a floating version of NuScale's SMR that could be deployed at sea close to shorelines.

In December 2022 NuScale announced it had completed the standard generic plant design for the VOYGR plant that will serve as a starting point for deploying site-specific designs.

Holtec SMR-160

Holtec International and its subsidiary SMR Inventec are developing a 160 MWe (525 MWt) factory-built reactor called the SMR-160. An integral pressurized light water reactor design with a single straight tube steam generator, the SMR-160 incorporates 57 uranium dioxide fuel assemblies with rod control assemblies and boron shim. The SMR-160 is passively cooled in operation and after shutdown for an indefinite period, with a negative temperature coefficient. The whole reactor system would be installed below ground level, with used fuel storage. A 24-month construction period is envisaged for each $600 million unit ($3750/kWe). The operational lifetime is at least 80 years.

The design passed the first phase of the Canadian Nuclear Safety Commission's (CNSC's) three-phase pre-licensing vendor design review in August 2020. Pre-licensing activities with the US Nuclear Regulatory Commission (NRC) are under way.

Holtec had earlier developed a concept design called the Holtec Inherently Safe Modular Underground Reactor (HI-SMUR). Pre-application discussions regarding the 145 MWe (469 MWt) design with the NRC took place at the end of 2010. The design had two external horizontal steam generators. The 32 full-length PWR fuel assemblies were in a fuel cartridge, which would be loaded and unloaded as a single unit from the 31-metre high pressure vessel.

Major revisions by 2012 led to the initial design of the SMR-160. The detailed design phase was from August 2012, and In March 2012 the US DOE signed an agreement with Holtec regarding the construction of a demonstration SMR-160 unit at its Savannah River Site in South Carolina. In 2013 NuHub, a South Carolina economic development project, and the state itself supported Holtec's bid for DOE funding for the SMR-160, as did partners PSEG and SCE&G – which would operate the demonstration plant – but DOE funding was eventually refused. However, in December 2020 the DOE selected Holtec for a $147.5 million development programme for the SMR-160 (DOE share $85.3 million under its Advanced Reactor Demonstration Program).

In August 2015 Mitsubishi Electric Power Products and its Japanese parent became a partner in the project, to undertake the digital instrumentation and control (I&C) design* and help with licensing. In January 2016 Holtec said that development continued with support from Mitsubishi and PSEG Power and in July 2017 a partner agreement with SNC-Lavalin based in Ontario was formalised, involving engineering support and licensing.

* All of Japan's PWRs and 14 Chinese PWRs use Mitsubishi Electric's I&C technology.

In 2017, Holtec began operation of a 500,000 sq ft (4.6 ha) weldment factory in Camden, NJ, designed to manufacture SMR components and equipment. The facility is currently manufacturing ASME pressure vessels and spent fuel storage and transport casks, and is capable of fabricating both SMR-160 and other SMR designs.

In April 2020 Holtec selected Framatome to supply its GAIA fuel assemblies for the reactor.

In November 2021 Holtec finalized an agreement with Hyundai Engineering & Construction of South Korea for the turnkey supply of the SMR-160 plant worldwide. Holtec will serve as the overall architect engineer for the plant and provide the major nuclear components through its US manufacturing facilities and international supply chain, and will provide the instrumentation and control systems through its partnership with Mitsubishi. Hyundai will contribute EPC and construction management capabilities for major projects.

In February 2019 Holtec announced new agreements with Exelon – to join the support team with Mitsubishi and SNC-Lavalin – and Ukraine's Energoatom, with which it had signed an agreement in 2018 with a view to building the SMR-160 in Ukraine. In June 2019 Holtec signed a partnership agreement with Energoatom and Ukraine's national nuclear consultant, State Scientific and Technical Centre for Nuclear and Radiation Safety (SSTC-NRS), to establish a consortium to explore the environmental and technical feasibility of qualifying a 'generic' SMR-160 system that can be built and operated at any candidate site in the country.

This would establish a reactor design capability in Ukraine, with a view to it becoming a regional hub for selling such reactors in Europe, Asia and Africa. In October 2020 Holtec signed an agreement with a subsidiary of Czech utility CEZ to evaluate deployment of the SMR-160 there.

In November 2021 Holtec said it aims to secure a US construction licence in 2025 and is "actively exploring the possibility" of deploying an SMR-160 at Oyster Creek – a decommissioning site which it acquired from Exelon in 2019 following the plant's closure – and at two other sites in southern USA.

mPower

In mid-2009, Babcock & Wilcox (B&W) announced its mPower reactor, a 500 MWt, 180 MWe integral PWR designed to be factory-made and railed to site[i]. It was a deliberately conservative design, to more readily gain acceptance and licensing. In November 2012 the US Department of Energy (DOE) announced that it would support accelerated development of the design for early deployment, with up to $226 million, and it paid $111 million of this.

The reactor pressure vessel containing core of 2x2 metres and steam generator is thus only 3.6 metres diameter and 22 m high, and the whole unit 4.5 m diameter and 23 m high. It would be installed below ground level, have an air-cooled condenser giving 31% thermal efficiency[p], and passive safety systems. The power was originally 125 MWe, but by about 2014, 195 MWe was quoted when water-cooled. A 155 MWe air-cooled version was also planned. The integral steam generator is derived from marine designs, as is the control rod set-up. Convection would be assisted by eight small canned-motor coolant pumps. It has a "conventional core and standard fuel" (69 fuel assemblies, each standard 17x17, < 20 t)[j] enriched to almost 5%, with burnable poisons, to give a four-year operating cycle between refuelling, which will involve replacing the entire core as a single cartridge. Core power density is lower than in a large PWR, and burn-up is about 35 GWd/t. (B&W draws upon over 50 years of experience in manufacturing nuclear propulsion systems for the US Navy, involving compact reactors with long core life.) A 60-year service life is envisaged, as sufficient used fuel storage would be built onsite for this.

The mPower reactor is modular in the sense that each unit is a factory-made module and several units would be combined into a power station of any size, but most likely a 380 MWe twin-unit plant and using approx 200 MWe turbine generators (also shipped as complete modules), constructed in three years. BWXT Nuclear Energy's present manufacturing capability in North America could produce these units.

B&W Nuclear Energy Inc set up B&W Modular Nuclear Energy LLC (now BWXT mPower Inc) to market the design, in collaboration with Bechtel which joined the project as a 10% equity partner to design, license and deploy it. The company expects both design certification and construction permit in 2018, and commercial operation of the first two units in 2022. Overnight cost for a twin-unit plant was put by B&W at about $5000/kW.

In November 2013 B&W said it would seek to bring in further equity partners by mid-2014 to take forward the licensing and construction of an initial plant.* B&W said it had invested $360 million in Generation mPower with Bechtel, and wanted to sell up to 70% of its stake in the joint venture, leaving it with about 20% and Bechtel 10%. In April 2014 B&W announced that it was cutting back funding on the project to about $15 million per year, having failed to find customers or investors. DOE then terminated further funding. B&W planned to retain the rights to manufacture the reactor module and nuclear fuel for the mPower plant. In December 2014 B&W finished laying off staff working on the project, and early in 2016 reduced funding further.

With more than $375 million having been spent on the mPower programme, in March 2016 BWXT and Bechtel reached agreement on “accelerated development” of the mPower project, so that Bechtel would take over leadership of the project and attempt for a year to secure funding for SMR development from third parties, including the DOE. If Bechtel succeeded in this, then BWXT and Bechtel would negotiate and execute a new agreement, with Bechtel taking over management of the mPower programme from BWXT. If Bechtel decided to terminate the project, it would be paid $30 million by BWXT, which is what happened in March 2017. The project was then shelved, leaving both BWXT and Bechtel free to be involved in the supply chain or management of other SMR projects.

* When B&W launched the mPower design in 2009, it said that Tennessee Valley Authority (TVA) would begin the process of evaluating Clinch River at Oak Ridge as a potential lead site for the mPower reactor, and that a memorandum of understanding had been signed by B&W, TVA and a consortium of regional municipal and cooperative utilities to explore the construction of a small fleet of mPower reactors. It was later reported that the other signatories of the agreement were FirstEnergy and Oglethorpe Power[3]. In February 2013 B&W signed an agreement with TVA to build up to four units at Clinch River, with design certification and construction permit application to be submitted to NRC in 2015. In August 2014 the TVA said it would file an early site permit (ESP) application instead of a construction permit application for one or more small modular reactors at Clinch River, possibly by the end of 2015. In February 2016 TVA said it was still developing a site at Oak Ridge for a SMR and would apply for an early site permit (ESP, with no technology identified) in May with a view to building up to 800 MWe of capacity there.

BWRX-300

GE Hitachi Nuclear Energy has a 300 MWe small BWR design, envisaged as single units. GEH has announced this as the BWRX-300 “which further simplifies the NRC-licensed ESBWR” from which it is derived. The BWRX-300 incorporates a range of cost-saving features, including natural circulation systems, smaller, dry containment, and more passive operational control systems. The estimated capital cost is $2250/kWe for series production after initial units are built. The design aims to limit onsite operational staff numbers to 75 employees to achieve an estimated O&M cost of $16/MWh. In May 2018 the US utility Dominion Energy agreed to help fund the project.

In July 2018 GEH announced $1.9 million in funding from the US Department of Energy to lead a team including Bechtel, Exelon, Hitachi-GE Nuclear Energy and the Massachusetts Institute of Technology to examine ways to simplify the reactor design, reduce plant construction costs, and lower operation and maintenance costs for the BWRX-300. In particular the team aims to identify ways to reduce plant completion costs by 40-60% compared with other SMR designs in development and to be competitive with gas. "As the tenth evolution of the boiling water reactor, the BWRX-300 represents the simplest, yet most innovative BWR design since GE began developing nuclear reactors in 1955." In May 2021 GEH said that if the design was selected by Ontario Power Generation it planned to bring the BWRX-300 to commercial readiness in partnership with OPG, and that it would be manufactured and constructed in Ontario, with the first unit built at Darlington. In October 2021 GEH engaged BWXT Canada for detailed engineering and design.

In May 2019 the BWRX-300 was submitted to Canada’s CNSC for a pre-licensing vendor design review. Phase 2 of this commenced in January 2020. After initiating discussion with the US Nuclear Regulatory Commission early in 2019, in January 2020 GE Hitachi announced it had submitted the first licensing topical report for the BWRX-300 SMR to the NRC, using the Part 50 two-step approach and leveraging the ESBWR design certification. GEH expects to have the first unit operating in the USA or Canada about 2028.

In October 2019 GEH signed an agreement with Estonia’s Fermi Energia and another agreement with Synthos SA in Poland to examine the economic feasibility of constructing a single BWRX-300 reactor in each country. In December 2020 Exelon in the USA completed a feasibility study for Synthos on deploying the BWRX-300. In June 2021 petrochemical company PKN Orlen joined Synthos in assessing the possibilities.

IRIS

Westinghouse's IRIS (International Reactor Innovative & Secure) is a reactor design which was developed over more than two decades. A 1000 MWt, 335 MWe capacity was proposed, although it could be scaled down to 100 MWe. IRIS is a modular pressurized water reactor with integral primary coolant system and circulation by convection. Fuel is similar to present LWRs and (at least for the 335 MWe version) fuel assemblies would be identical to those in AP1000. Enrichment is 5% with burnable poison and fuelling interval of up to four years (or longer with higher enrichment and MOX fuel). US design certification was at the pre-application stage, but is now listed as 'inactive', and the concept has evolved into the Westinghouse SMR.

Westinghouse SMR

The Westinghouse small modular reactor is an 800 MWt/225 MWe class integral PWR with passive safety systems and reactor internals including fuel assemblies based closely on those in the AP1000 (89 assemblies 2.44m active length, <5% enrichment). The steam generator is above the core fed by eight horizontally-mounted axial-flow coolant pumps. The reactor vessel will be factory-made and shipped to site by rail, then installed below ground level in a containment vessel 9.8 m diameter and 27 m high. The reactor vessel module is 25 metres high and 3.5 metres diameter. It has a 24-month refueling cycle and 60-year service life. Passive safety means no operator intervention is required for seven days in the event of an accident. Daily load following can be performed from 100% to 20% power at a rate of 5% change per minute; in continuous load following, the plant can perform load changes of ±10% power at a rate of 2% per minute.

In May 2012 Westinghouse teamed up with General Dynamics Electric Boat to assist in the design and Burns & McDonnell to provide architectural and engineering support. A design certification application was expected by NRC in September 2013, but the company has stepped back from lodging one while it re-assesses the market for small reactors. The company has started fabricating prototype fuel assemblies.

The DOE earlier saw this as a "near-term LWR design". In March 2015 Westinghouse announced that the NRC had approved its safety evaluation report for the SMR design, which it said was a significant step towards design certification. However, while the company continues efforts to seek customer interest, it is not proceeding with the NRC yet.

In April 2012 Westinghouse set up a project with Ameren Missouri to seek DOE funds for developing the design, with a view to obtaining design certification and a combined construction and operation licence (COL) from the Nuclear Regulatory Commission (NRC) for up to five SMRs at Ameren's Callaway site, instead of an earlier proposed large EPR there. The initiative – NexStart SMR Alliance – had the support of other state utilities and the state governor, as well as Savannah River, Exelon and Dominion. However, this agreement expired about the end of 2013, and both companies stepped back from the project as DOE funds went to other SMR projects.

In May 2013 Westinghouse announced that it would work with China's State Nuclear Power Technology Corporation (SNPTC) to accelerate design development and licensing in the USA and China of its SMR. SNPTC would ensure that the Westinghouse SMR design met standards for licensing in China and would lead the licensing effort in that country. The status of this collaboration is uncertain.

In October 2015 Westinghouse presented a proposal for a "shared design and development model" under which the company would contribute its SMR conceptual design and then partner with UK government and industry to complete, license and deploy it. This would engage UK companies such as Sheffield Forgemasters in the reactor supply chain.

VVER-300 (V-478)

This is a 850 MWt, 300 MWe two-loop PWR design from Gidropress, based on the VVER-640 (V-407) design. It is little reported.

VBER-150, VBER-300

A larger Russian factory-built and barge-mounted unit (requiring a 12,000 tonne vessel) is the VBER-150, of 350 MWt, 110 MWe. It is modular and is derived by OKBM from naval designs, with two steam generators. Uranium oxide fuel enriched to 4.7% has burnable poison; it has low burn-up (31 GWd/t average, 41.6 GWd/t maximum) and eight-year refuelling interval.

OKBM Afrikantov's larger VBER-300 PWR is a 917 MWt, 325 MWe unit, the first of which is planned to be built in Kazakhstan. It was originally envisaged in pairs as a floating nuclear power plant, displacing 49,000 tonnes. As a cogeneration plant it is rated at 200 MWe and 1900 GJ/hr. The reactor is designed for 60-year life and 90% capacity factor. It has four external steam generators and a cassette core with 85 standard VVER fuel assemblies enriched to 4.95% and 50 GWd/tU burn-up with a 72-month fuel cycle. Versions with three and two steam generators are also envisaged, of 230 and 150 MWe respectively. Also, with more sophisticated and higher-enriched (18%) fuel in the core, the refuelling interval can be pushed from two years out to five years (6 to 15 years fuel cycle) with burn-up to 125 GWd/tU. A 2006 joint venture between Atomstroyexport and Kazatomprom set this up for development as a basic power source in Kazakhstan, then for export[e]. It is also envisaged for use in Russia, mainly as cogeneration unit. It is considered likely for near-term deployment.

The company also offers 200-600 MWe designs based on a standard 100 MWe module and explicitly based on naval units.

VK-300

Another larger Russian reactor with completed detailed design is NIKIET's VK-300 integral boiling water reactor of 750 MWt, 250 MWe, being developed specifically for cogeneration of both power and district heating or heat for desalination (150 MWe plus 1675 GJ/hr) by the N.A. Dollezhal Research and Development Institute of Power Engineering (RDIPE or NIKIET) together with several major research and engineering institutes. It has evolved from the 50 MWe (net) VK-50 BWR at Dimitrovgrad[f], but uses standard components wherever possible, and has 313 fuel elements similar to the VVER. Cooling is passive, by convection, and all safety systems are passive. Fuel enrichment is 4% and burn-up is 41 GWd/tU with a 72-month refuelling interval. It is capable of producing 250 MWe if solely electrical. Design operating lifetime is 60 years.

In September 2007 it was announced that six would be built at Kola or Archangelsk and at Primorskaya in the far east, to start operating 2017-20,[4] but no more has been heard of this plan. A feasibility study was undertaken for Arkhangelsk nuclear cogeneration plant with four units. As a cogeneration plant it was intended for the Mining & Chemical Combine at Zheleznogorsk, but MCC is reported to prefer the VBER-300. The design was completed in 2013.

VKT-12

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

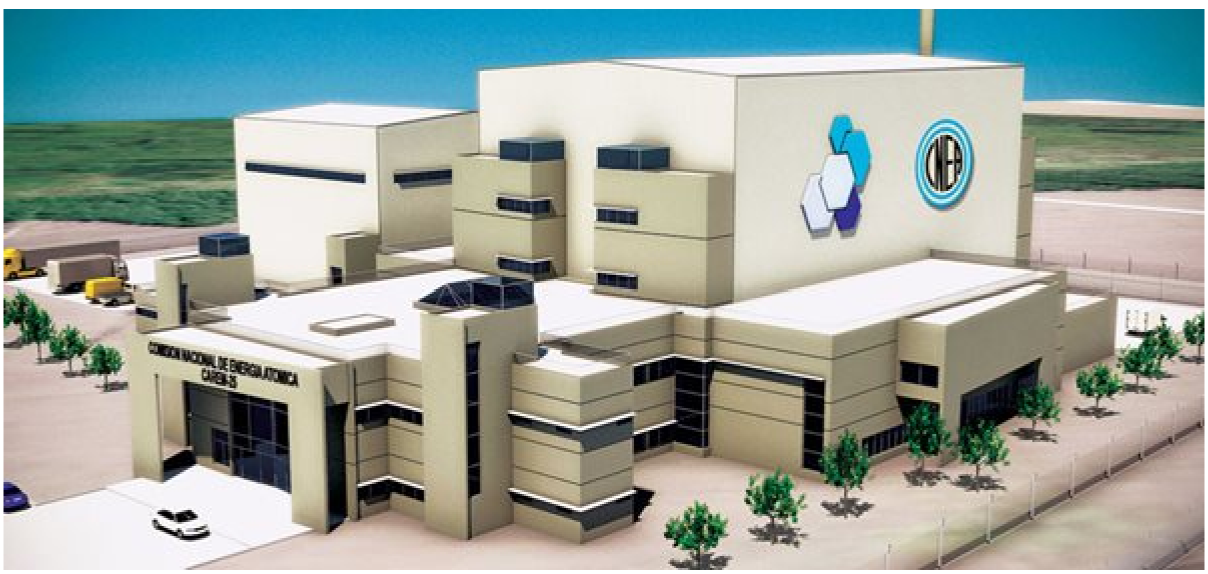
ABV, ABV-6M

A smaller Russian PWR unit under development by OKBM Afrikantov is the ABV multipurpose power source. It is readily transported to the site, with rapid assembly and operation for 10-12 years between refuelling, which is carried out offsite at special facilities. There is a range of sizes from 45 MWt (ABV-6M) down to 18 MWt (ABV-3), giving 4-18 MWe outputs. (The IAEA 2011 write-up of the ABV-6M quotes 14 MWt or 6 MWe in cogeneration mode.) The units are compact, with integral steam generator and natural circulation in the primary circuit. They will be factory-produced and designed as a universal power source for floating nuclear plants – the ABV-6M would require a 3500 tonne barge; the ABV-3, 1600 tonne for twin units. The Volnolom FNPP consists of a pair of reactors (12 MWe in total) mounted on a 97-metre, 8700 tonne barge plus a second barge for reverse osmosis desalination (over 40,000 m^3/day of potable water).

The smallest land-based version has reactor module 13 m long and 8.5 m diameter, with a mass of 600 t. The land-based ABV-6M module is 44 m long, 10 m diameter and with mass of 3000 t. The core is similar to that of the KLT-40 except that enrichment is 16.5% or 19.7% and average burn-up 95 GWd/t. It would initially be fuelled in the factory. The service lifetime is about 40 years.

CAREM

The CAREM25 reactor prototype being built by the Argentine National Atomic Energy Commission (CNEA), with considerable input from INVAP[g], is an older design modular 100 MWt (27 MWe gross) integral pressurized water reactor, first announced in 1984. It has 12 steam generators within the pressure vessel and is designed to be used for electricity generation or as a research reactor or for water desalination (with 8 MWe in cogeneration configuration). CAREM has its entire primary coolant system within the reactor pressure vessel (11m high, 3.5m diameter), self-pressurized and relying entirely on convection (for modules less than 150 MWe). The final full-sized export version will be 100 MWe or more, with axial coolant pumps driven electrically. Fuel is standard 3.1 or 3.4% enriched PWR fuel in hexagonal fuel assemblies, with burnable poison, and is refuelled annually.



How a CAREM plant would look (CNEA)

The 25 MWe prototype unit is being built next to Atucha, on the Parana River in Lima, 110 km northwest of Buenos Aires, and the first larger version (probably 100 MWe) is planned in the northern Formosa province, 500 km north of Buenos Aries, once the design is proven. Some 70% of CAREM25 components will be local manufacture. The pressure vessel is being manufactured by Industrias Metalurgicas Pescarmona SA (IMPSA).

The IAEA lists it as a research reactor under construction since April 2013, though first concrete was poured in February 2014. It is proceeding slowly and was originally due online in 2019.

In March 2015 Argentina's INVAP and state-owned Saudi technology innovation company Taqnia set up a joint venture company, Invania, to develop nuclear technology for Saudi Arabia's nuclear power programme, apparently focused on CAREM for desalination.

SMART from KAERI, Korean SMR

On a larger scale, South Korea's SMART (System-integrated Modular Advanced Reactor) is a 330 MWt pressurized water reactor with integral steam generators and advanced safety features. It is designed by the Korea Atomic Energy Research Institute (KAERI) for generating electricity (up to 100 MWe) and/or thermal applications such as seawater desalination. Design operating lifetime is 60 years, fuel enrichment 4.8%, with a three-year refuelling cycle. It has 57 fuel assemblies very similar to normal PWR ones but shorter, and it operates with a 36-month fuel cycle. All the active safety features of the original design were substituted by early 2016 with passive versions. Residual heat removal is passive. It received standard design approval (SDA) from the Korean regulator in mid-2012. A single unit can produce 90 MWe plus 40,000 m^3/day of desalinated water.

In March 2015 KAERI signed an agreement with Saudi Arabia's King Abdullah City for Atomic and Renewable Energy (KA-CARE) to assess the potential for building SMART reactors in that country, and in September 2015 further contracts were signed to that end. The cost of building the first SMART unit in Saudi Arabia was estimated at $1 billion. Through to November 2018 pre-project engineering was to be undertaken jointly including FOAK engineering design and preparations for building two units.

In April 2021 Korea Hydro & Nuclear Power (KHNP) announced that it was working with KAERI to improve the economics of the SMART design, with an aim of obtaining a licence for a new Korean SMR of 170 MWe with good load-following ability by 2028, with a view to exports.

BANDI-60S

The BANDI-60S is a two-loop PWR being developed since 2016 by South Korea's Kepco Engineering & Construction company. It is a 200 MWt/60 MWe reactor designed for niche markets, particularly floating nuclear power plants. It is described as 'block type' with the external steam generators connected directly nozzle-to-nozzle. Initially the steam generators are conventional U-tube, but Kepco is working on a plate and shell design which will greatly reduce their size. Apart from steam generators, most main components including control rod drives are within the pressure vessel. Primary pumps are canned motor, and decay heat removal is passive. There are 52 conventional fuel assemblies, giving 35 GWd/t burn-up with 48-60 month fuel cycle. Burnable absorbers are used instead of soluble boron. Design operating lifetime is 60 years. The pressure vessel is 11.2 m high and 2.8 m diameter. In September 2020 Kepco signed an agreement with Daewoo Shipbuilding & Engineering to develop offshore nuclear power plants using the reactor.

MRX

The Japan Atomic Energy Research Institute (JAERI) designed the MRX, a small (50-300 MWt) integral PWR reactor for marine propulsion or local energy supply (30 MWe). The entire plant would be factory-built. It has conventional 4.3% enriched PWR uranium oxide fuel with a 3.5-year refuelling interval and has a water-filled containment to enhance safety. Little has been heard of it since the start of the Millennium.

Nuward NP-300

TechnicAtome with Naval Group and CEA in France have developed the NP-300 PWR design from naval power plants and aimed it at export markets for power, heat and desalination. It is a PWR with passive safety systems and could be built for applications of 100 to 300 MWe or more with up to 500,000 m^3/day desalination. As of mid-2018, a 570 MWt/170 MWe version was proposed, in a metallic compact containment submerged in water. In September 2019 twin 170 MWe units were proposed to comprise a 340 MWe power plant, with two reactors sharing a pool. A partnership with Westinghouse was being considered. EdF plans to enter the basic design pre-licensing phase with ASN in 2022. Some €1 billion state funding is promised for the project.

EDF is "targeting replacing ageing coal plants of the 300 to 400 MW range" with two-unit Nuward plants, as well as at supplying remote municipalities and energy intensive industrial sites and powering small grids.

TechnicAtome makes the K15 naval reactor of 150 MWt, running on low-enriched fuel. A land-based equivalent – *Réacteur d'essais à terre* (RES) – was built at Cadarache from 2003 with several delays and achieved criticality in October 2018. It is essentially a PWR test reactor for the Navy.

It earlier seemed that some version of this reactor might be used in the Flexblue submerged nuclear power plant being proposed by DCNS in France, but now cancelled. The concept eliminated the need for civil engineering, and refuelling or major service could be undertaken by refloating it and returning to the shipyard.

NHR-200

The Chinese NHR-200 (Nuclear Heating Reactor), developed by Tsingua University's Institute of Nuclear Energy Technology (now the Institute of Nuclear and New Energy Technology), is a simple 200 MWt integral PWR design for district heating or desalination. It is based on the NHR-5 which was commissioned in 1989, and heated the INET campus for three winters[h].

It has convection circulation at 2.5 MPa in primary circuit pressure to produce steam at 127°C. Used fuel is stored around the core in the pressure vessel. The first NHR-200 plants are proposed for Daqing city in Heilongjiang province and Shenyang in Liaoning province.

The NHR200-II with design and verification tests concluded in 2016 operates at 8 MPa primary circuit pressure to produce steam at over 200°C and can also be used for power generation, seawater desalination or heat for mineral processing.

ACP100/Linglong One

The Nuclear Power Institute of China (NPIC), under China National Nuclear Corporation (CNNC), has designed a multi-purpose small modular reactor, the ACP100 or Linglong One. It has passive safety features, notably decay heat removal, and will be installed underground. Seismic tolerance is 300 Gal. It has 57 fuel assemblies 2.15m tall and integral steam generators (320°C), so that the whole steam supply system is produced and shipped a single reactor module. Its 385 MWt produces about 125 MWe, and power plants comprising two to six of these are envisaged, with 60-year design operating lifetime and 24-month refuelling. Or each module can supply 1000 GJ/hr, giving 12,000 m^3/day desalination (with MED). Industrial and district heat uses are also envisaged, as well as floating nuclear power plant (FNPP) applications. Capacity of up to 150 MWe is envisaged. In April 2016 the IAEA presented CNNC with its report from the Generic Reactor Safety Review process.

In October 2015 the Nuclear Power Institute of China (NPIC) signed an agreement with UK-based Lloyd's Register to support the development of a floating nuclear power plant (FNPP) using the ACP100S reactor, a marine version of the ACP100. Following approval as part of the 13th Five-Year Plan for innovative energy technologies, CNNC signed an agreement in July 2016 with China Shipbuilding Industry Corporation (CSIC) to prepare for building its ACP100S demonstration floating nuclear plant.

The Linglong One Demonstration Project* at Changjiang on Hainan Island involves a joint venture of three main companies: CNNC as owner and operator; the Nuclear Power Institute of China (NPIC) as the reactor designer; and China Nuclear Power Engineering Group (CNPE) being responsible for plant construction. The preliminary safety analysis report for a single unit demonstration plant was approved in April 2020. In May 2022 pouring of concrete for the reactor's basemat was completed. Construction time is expected to be 58 months.

* Hainan Changjiang Multi-purpose Small Modular Reactor Technical Demonstration Project is the full name.

CNNC signed a second ACP100 agreement with Hengfeng county, Shangrao city in Jiangxi province, and a third with Ningdu county, Ganzhou city in Jiangxi province in July 2013 for another ACP100 project costing CNY 16 billion. Further inland units are planned in Hunan and possibly Jilin provinces. Export potential is considered to be high, with full IP rights. In 2016 CNPE submitted an expression of interest to the UK government based on its ACP100+ design.

CAP200/LandStar-V, CAP150, CAP50, LandStar-I

CAP200 or LandStar-V multiple application SMR is a PWR, with SNPTC provenance, being developed from the CAP1000 in parallel with the CAP1400 by SNERDI, using proven fuel and core design. It is 660 MWt/220 MWe and has two external steam generators (301°C). It is pitched to replace coal plants and supply process heat and district heating, with a design operating lifetime of 60 years. With 24-month refuelling, burn-up of 42 GWd/t is expected, the 89 fuel assemblies being the same as those of the CAP1400 but shorter. It has both active and passive cooling, and natural circulation is effective for up to 20% power. In an accident scenario, no operator intervention is required for seven days. It will be installed below grade in a 32 m deep caisson structure, with seismic design basis 600 Gal, even in soft ground. In 2017 the first-of-a-kind cost was estimated at $5000/kW and $160/MWh, dropping to $4000/kW in series.

The OceanStar-V version would be on a barge, as a floating nuclear power plant.

The CAP150 is an earlier version, 450 MWt/150 MWe, with eight integral steam generators. It is claimed to have “a more simplified system and more safety than current third generation reactors.” Seismic design basis is 300 Gal. In mid-2013 SNPTC quoted approximately $5000/kW capital cost and 9 c/kWh, so significantly more than the CAP1400.

A related SNERDI project is the CAP50 reactor for floating nuclear power plants. This is to be 200 MWt and relatively low-temperature (250°C), so only about 40 MWe with two external steam generators and five-year refuelling.

SPIC’s LandStar-I is an integral pressure-vessel reactor of 200 MWt with convection circulation at 9 MPa producing hot water for district heating. At SPIC’s Jiamusi demonstration project in Heilongjiang province, two 200 MW LS-I reactors are being built.

ACPR100, ACPR50S

China General Nuclear Group (CGN) has two small ACPR designs: an ACPR100 and ACPR50S, both with passive cooling for decay heat and 60-year design operating lifetime. Both have standard type fuel assemblies and fuel enriched to <5% with burnable poison giving 30-month refuelling. The ACPR100 is an integral PWR, 450 MWt, 140 MWe, having 69 fuel assemblies. Reactor pressure vessel is 17m high and 4.4 m inside diameter, operating at 310 °C. It is designed as a module in larger plant and would be installed underground. The applications for these are similar to those for the ACP100.

CGN's floating reactor concept

The offshore ACPR50S is 200 MWt, 60 MWe with 37 fuel assemblies and four external steam generators. Reactor pressure vessel is 7.4m high and 2.5 m inside diameter, operating at 310 °C. It is designed for mounting on a barge as a floating nuclear power plant (FNPP). Following approval as part of the 13th Five-Year Plan for innovative energy technologies, CGN announced the construction start on the first FNPP at Bohai Shipyard in Huludao, southwestern Liaoning province, in November 2016. No further announcements on the project have since been made.

HHP25

China Shipbuilding Industry Corporation (CSIC) is developing FNPPs powered by 100 MWt (25 MWe) HHP25 reactors, derived from a submarine reactor by CSIC's No. 719 Research Institute. At the Dalian Maritime Exhibition in October 2018, CSIC said the "offshore nuclear power platform" would be 163 m long, 29 m wide with a displacement of 29,800 t. It is powered by two HHP25 reactors and can supply up to 200 t/d of desalinated water.

Flexblue

This was a conceptual design from DCNS (now Naval Group, state-owned), Areva, EdF and CEA from France. It is designed to be submerged, 60-100 metres deep on the sea bed up to 15 km offshore, and returned to a dry dock for servicing. The reactor, steam generators and turbine-generator would be housed in a submerged 12,000 tonne cylindrical hull about 100 metres long and 12-15 metres diameter. Each hull and power plant would be transportable using a purpose-built vessel. Reactor capacity ranged 50-250 MWe, derived from DCNS's latest naval designs, but with details not announced. In 2011 DCNS said it could start building a prototype Flexblue unit in 2013 in its shipyard at Cherbourg for launch and deployment in 2016, possibly off Flamanville, but the project has been cancelled.

UNITHERM

This is an integral 30 MWt, 6.6 MWe PWR conceptual design from Russia's Research and Development Institute of Power Engineering (RDIPE or NIKIET). It has three coolant loops, with natural circulation, and claims self-regulation with burnable poisons in unusual metal-ceramic fuel design, so needs no more than an annual maintenance campaign and no refueling during a 25-year life. The mass of one unit with shielding is 180 tonnes, so it can be shipped complete from the factory to site.

SHELF

This is a Russian 6 or 10 MWe, 28 MWt integral PWR concept with turbogenerator in a cylindrical pod about 15 m long and 8 m diameter, sitting on the sea bed like Flexblue. The SHELF module uses an integral reactor with forced and natural circulation in the primary circuit, in which the core, steam generator, motor-driven circulation pump and control and protection system drive are housed in a cylindrical pressure vessel. It uses low-enriched fuel of UO_2 in aluminium alloy matrix. Fuel cycle is 56 months. The reactor is based on operating prototypes, and would be serviced infrequently. It is intended as energy supply for oil and gas developments in Arctic seas, and land-based versions have been envisaged. It is at concept design stage with NIKIET which estimates that a further five years would be required in order to finalize the design, licensing, construction and commissioning. Completion of the technical design is envisaged in 2024.

KARAT-45

This is a 45 MWe tank-type BWR as a stand-alone cogeneration plant. The design includes natural circulation in its core cooling system for heat removal in all operational modes and incorporates passive safety systems. A larger version is 100 MWe.

IMR

Mitsubishi Heavy Industries has a conceptual design of the Integral Modular Reactor (IMR), a PWR of 1000 MWt, 350 MWe. It has design operating lifetime of 60 years, 4.8% fuel enrichment and fuel cycle of 26 months. It has natural circulation for primary cooling. The project has involved Kyoto University, the Central Research Institute of the Electric Power Industry (CRIEPI), and the Japan Atomic Power Company (JAPC), with funding from METI. The target year to start licensing was 2020 at the earliest, but the design appears to have been dropped.

Rolls-Royce SMR

Rolls-Royce has been working since 2015 on a design that was originally 220 MWe, but the focus has changed to a medium-sized reactor of 400-440 MWe (1200-1350 MWt), and from 2021 was referred to as "at least 470 MW". It is a three-loop PWR with close-coupled external steam generators. It is to be factory-built, with major components transportable to site (RPV: 11.3 m high, 4.5 m diameter, SG: 4.95 m diameter, about 25 m high) and assembled in 500 days. It has a 60-year design operating lifetime. It would use 4.95% enriched fuel with 55-60 GWd/t burn-up in 121 standard PWR fuel assemblies with active fuelled length of 2.8 m and using burnable poison in 40 out of 264 fuel rods in each. The refuelling cycle would be 18-24 months. One such unit would comprise a stand-alone power plant.

Early in 2016 Rolls-Royce submitted a detailed design to the UK government for a 220 MWe SMR unit and also a paper to the Department of Business, Energy and Industrial Strategy, outlining its plan to develop a fleet of 7 GWe of SMRs in the UK with a new consortium, plus 9 GWe of exported units. In 2020 the partners with Rolls-Royce were: Assystem, Atkins, BAM Nuttall, Laing O'Rourke, National Nuclear Laboratory, Nuclear AMRC, Jacobs and The Welding Institute; and in November 2020 it added US utility Exelon with a view to it operating Rolls-Royce SMRs in the UK and abroad. Its focus is on existing licensed nuclear sites in the UK, notably Trawsfynydd in north Wales, the site of a former Magnox nuclear power station. It is hoping to have the first unit operating in 2030.

In May 2021 the cost of a 470 MWe unit was put at about £1.8 billion, so $5100/kW, and levelized cost of electricity (LCOE) at £35-50/MWh. The company submitted the design for the UK generic design assessment (GDA) process in November 2021, and in March 2022 the ONR began the GDA.

In November 2017, Rolls-Royce signed a memorandum of understanding (MoU) with the Jordan Atomic Energy Commission to conduct a technical feasibility study for the construction of a Rolls-Royce SMR in the Middle Eastern country. In March 2020, Turkey's state-owned EUAS International ICC signed an MoU with Rolls-Royce to evaluate the technical, economical and legal applicability of SMRs. In addition, the companies will consider the possibility of joint production of such reactors. In November 2020 Rolls-Royce announced an agreement with Czech utility CEZ to assess potential deployment there.

Rolls-Royce has designed three generations of naval reactors since the 1950s and also operates a small test reactor. It led the design of a small integral reactor (SIR) of 330 MWe in the late 1980s.

TRIGA

The TRIGA Power System is a PWR concept based on General Atomics' well-proven research reactor design. It is conceived as a 64 MWt, 16.4 MWe pool-type system operating at a relatively low temperature. The secondary coolant is perfluorocarbon. The fuel is uranium-zirconium hydride enriched to 20% and with a little burnable poison and requiring refuelling every 18 months.

Used fuel is stored inside the reactor vessel.

FBNR

The Fixed Bed Nuclear Reactor (FBNR) is an early conceptual design from the Federal University of Rio Grande do Sul, Brazil. It is an integral PWR of 218 MWt, 70 MWe, with 15 mm pebble fuel.

The reactor consists of an active core (1.7 m diameter, 2 m height) and integral upper steam generator within a 6 m high vessel, and a fuel chamber located beneath the core. The fuel is carried up from the fuel chamber into the core by the coolant, which absorbs the core heat and continues into the steam generator. The coolant then returns to the fuel chamber via the coolant pump, forming a closed loop. Cutting the power to the pump shuts down the reactor by causing the fuel pebbles to fall from the core into the fuel chamber.

The Triso fuel particles comprise 5% enriched 0.5 mm diameter UO_2 fuel kernels within a single 0.1 mm thick carbon shell. Each 15 mm fuel pebble consists of fuel particles within a silicon carbide matrix (60% fuel and 40% SiC) enclosed in a 0.5 mm thick stainless steel outer layer.

SMART from Dunedin

The SMART (Small Modular Adaptable Reactor Technology) from Dunedin Energy Systems in Canada is a 30 MWt, 6 MWe battery-type unit, installed below grade. It is replaced by a new one when it is returned to a processing facility for refuelling; at 83% capacity factor this would be every 20 years. It drives a steam turbine. Emergency cooling is by convection. Cost is about 29c/kWh, according to Dunedin.

DEER from Radix

The DEER (Deployable Electric Energy Reactor) was being developed by Radix Power & Energy Corporation in the USA, in collaboration with Brookhaven Technology Group, Brookhaven National Laboratory, Parsons Corporation, Dunedin Energy Systems, and University of California, Berkeley. The DEER is a PWR and would be portable and sealed, able to operate in the range of 10-50 MWe. DEER-1 was to use fuel based on that in Triga research reactors, with a ten-year cycle, and DEER-2 was to use TRISO fuel, for forward military bases or remote mining sites. No recent information is available.

Chinese district heat reactors

Three Chinese designs are solely for district heat at 90-110°C, for northern provinces, especially Heilongjiang. Reducing winter air pollution is the main driver of their development. CGN’s NHR-200 passed regulatory review in the 1990s; CNNC’s DHR-400 or 'Yanlong' is a 400 MWt pool-type reactor; and SPIC’s LandStar-I is similar to the Yanlong but 200 MWt.

Heavy water reactors

PHWR-220

These are the oldest and smallest of the Indian pressurized heavy water reactor (PHWR) range, with a total of 16 now online, 800 MWt, 220 MWe gross typically. Rajasthan 1 was built as a collaborative venture between Atomic Energy of Canada Ltd (AECL) and the Nuclear Power Corporation of India (NPCIL), starting up in 1972. Subsequent indigenous PHWR development has been based on these units, though several stages of evolution can be identified: PHWRs with dousing and single containment at Rajasthan 1&2, PHWRs with suppression pool and partial double containment at Madras, and later standardized PHWRs from Narora onwards having double containment, suppression pool, and calandria filled with heavy water, housed in a water-filled calandria vault. They are moderated and cooled by heavy water, and the natural uranium oxide fuel is in horizontal pressure tubes, allowing refuelling online (maintenance outages are scheduled after 24 months). Burn-up is about 15 GWd/t.

AHWR-300 LEU

The Advanced Heavy Water Reactor developed by the Bhaba Atomic Research Centre (BARC) is designed to make extensive use of India’s abundant thorium as fuel, but a low-enriched uranium fuelled version is pitched for export. This will use low-enriched uranium plus thorium as a fuel, largely dispensing with the plutonium input of the version for domestic use. About 39% of the power will come from thorium (via in situ conversion to U-233, cf two-thirds in domestic AHWR), 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. It will have vertical pressure tubes in which the light water coolant under high pressure will boil, circulation being by convection. Nominal 300 MWe, 284 MWe net. It is at the basic design stage.

High-temperature gas-cooled reactors

These use graphite as moderator (unless fast neutron type) and either helium, carbon dioxide or nitrogen as primary coolant. The experience of several innovative reactors built in the 1960s and 1970s[k], notably those in Germany, has been analyzed, especially in the light of US plans for its Next Generation Nuclear Plant (NGNP) and China's launching its HTR-PM project in 2011. Lessons learned and documented for NGNP include the use of TRISO fuel, use of a reactor pressure vessel, and use of helium cooling (UK AGRs are the only HTRs to use CO2 as primary coolant). However US government funding for NGNP has now virtually ceased, and the technology lead has passed to China.

New high-temperature gas-cooled reactors (HTRs) are being developed which will be capable of delivering high temperature (700-950ºC and eventually up to about 1000°C) helium either for industrial application via a heat exchanger, or to make steam conventionally in a secondary circuit via a steam generator, or directly to drive a Brayton cycle* gas turbine for electricity with almost 50% thermal efficiency possible (efficiency increases around 1.5% with each 50°C increment). One design uses the helium to drive an air compressor to supercharge a CCGT unit. Improved metallurgy and technology developed in the last decade makes HTRs more practical than in the past, though the direct cycle means that there must be high integrity of fuel and reactor components. All but one of those described below have neutron moderation by graphite, one is a fast neutron reactor.

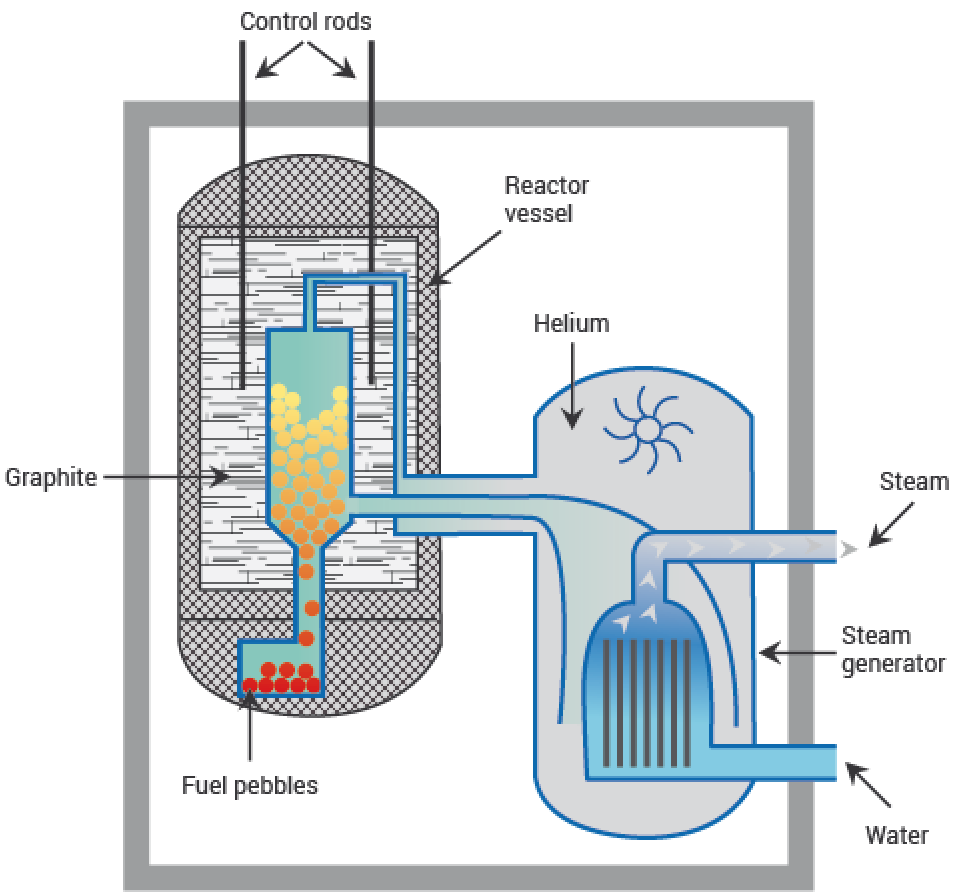
* There is little interest in pursuing the direct Brayton cycle for helium at present due to higher technological risk. Attrition of fuel tends to give rise to graphite dust with radioactivity in the coolant circuit. Also the helium needs to be very pure to avoid corrosion.

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

There are two ways in which these particles are arranged: in blocks – hexagonal 'prisms' of graphite, or in billiard ball-sized pebbles of graphite, each with about 15,000 fuel particles and 9g uranium. There is a greater volume of used fuel (20 times) than from the same capacity in a light water reactor, due to the fact that the fuel pebbles are mainly graphite – less than one percent is uranium. However, the used fuel is overall less radiotoxic and produces less decay heat due to higher burn-up. The HTR moderator is graphite.

There are several designs for gas-cooled fast reactors, mostly large. One small design is General Atomics EM^2, with helium cooling. Another – th supercritical direct cycle gas fast reactor – is based on the UK’s AGR, cooled by carbon dioxide. Both are described below.

A High-Temperature Reactor (HTR)



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

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

Three HTR designs in particular – PBMR, GT-MHR and Areva's SC-HTGR – were contenders for the Next Generation Nuclear Plant (NGNP) project in the USA (see *Next Generation Nuclear Plant* section in the information page on US Nuclear Power Policy). In 2012 Areva's HTR was chosen. However, the only commercial-scale HTR project currently proceeding is the Chinese HTR-PM.

Hybrid Power Technologies have a hybrid-nuclear Small Modular Reactor (SMR) coupled to a fossil-fuel powered gas turbine.

HTTR, GTHTR-300C, HTR50S

Japan Atomic Energy Agency's (previously Japan Atomic Energy Research Institute's) High-Temperature Test Reactor (HTTR) of 30 MWt started up at the end of 1998 and first reached full power with a reactor outlet coolant temperature of 850°C in December 2001. In 2004 it achieved 950°C outlet temperature, and in 2009 it ran at 950°C for 50 days. Its fuel is TRISO particles with low-enriched (average 6%) uranium in prisms and its main purpose is to develop a thermochemical means of producing hydrogen from water.

Based on the HTTR, JAERI is developing the Gas Turbine High Temperature Reactor 300 for Cogeneration (GTHTR-300C) of up to 600 MWt per module. It uses improved HTTR fuel elements with 14% enriched uranium achieving high burn-up (120 GWd/t). Helium at 850-950°C drives a horizontal turbine at 47% efficiency to produce up to 300 MWe. The core consists of 90 hexagonal fuel columns 8 metres high arranged in a ring, with reflectors. Each column consists of eight one-metre high elements 0.4 m across and holding 57 fuel pins made up of fuel particles with 0.55 mm diameter kernels and 0.14 mm buffer layer. In each two-yearly refuelling, alternate layers of elements are replaced so that each remains for four years. It is being developed with Mitsubishi Heavy Industries (MHI), Toshiba/IHI and Fuji, and target for commercialization is about 2030.

JAEA's small HTR50S reactor based on the HTTR is a conceptual design for industrial process and heat and/or power generation. This is 50 MWt with dual reactor outlet temperatures of 750°C and 900°C with maximum use of conventional technologies in order to deploy them in developing countries in the 2020s. Initially this would use a steam cycle for power generation, then improve the fuel, and then Increase the reactor outlet temperature to 900°C and install an intermediate heat exchanger (IHX) to demonstrate helium GT and hydrogen production using the IS process.

Early in 2019 the Japan Atomic Energy Agency (JAEA) formed a joint venture with Penultimate Power UK to build a 10 MWe SMR there based on the HTTR – referred to as the EH HTGR – for power and process heat in industrial clusters. Plans include scaling up the design to 100 MWe and building a factory in the UK for multiple plants. Penultimate Power claims the first reactor will cost about £100 million ($140 million), with reductions for future units. It expects the first reactor to be operating by 2029.

HTR-10

China's HTR-10, a 10 MWt high-temperature gas-cooled experimental reactor at the Institute of Nuclear & New Energy Technology (INET) at Tsinghua University north of Beijing started up in 2000 and reached full power in 2003. It has its fuel as a 'pebble bed' (27,000 elements) of oxide fuel with average burn-up of 80 GWday/tU. Each pebble fuel element has 5g of uranium enriched to 17% in around 8300 TRISO-coated particles. The reactor operates at 700°C (potentially 900°C) and has broad research purposes. Eventually it will be coupled to a gas turbine, but meanwhile it has been driving a steam turbine.

In 2004, the small HTR-10 reactor was subject to an extreme test of its safety when the helium circulator was deliberately shut off without the reactor being shut down. The temperature increased steadily, but the physics of the fuel meant that the reaction progressively diminished and eventually died away over three hours. At this stage a balance between decay heat in the core and heat dissipation through the steel reactor wall was achieved, the temperature never exceeded a safe 1600°C, and there was no fuel failure. This was one of six safety demonstration tests conducted then. The high surface area relative to volume, and the low power density in the core, will also be features of the full-scale units (which are nevertheless much smaller than most light water types.)

HTR-PM

Construction of a larger version of the HTR-10, China's HTR-PM, was approved in principle in November 2005, with preparation for first concrete in mid-2011 and full construction start intended in December 2012. It is also based on the German HTR-Modul design of 200 MWt. Originally envisaged as a single 200 MWe (450 MWt) unit, this will now have twin reactors, each of 250 MWt driving a single 210 MWe steam turbine.*

* The size was reduced to 250 MWt from earlier 458 MWt modules in order to retain the same core configuration as the prototype HTR-10 and avoid moving to an annular design like South Africa's PBMR (see section on PMBR below)

Each reactor has a single steam generator with 19 elements (665 tubes). The fuel as 60 mm diameter pebbles is 8.5% enriched (520,000 elements in the two reactors) giving 90 GWd/t discharge burn-up. Core outlet temperature is 750ºC for the helium, steam temperature is 566°C and core inlet temperature is 250°C. It has a thermal efficiency of 40%. Core height is 11 metres, diameter 3 m in a 25 m high, 5.7 m diameter reactor vessel. There are two independent reactivity control systems: the primary one consists of 24 control rods in the side graphite reflector, the secondary one of six channels for small absorber spheres falling by gravity, also in the side reflector. Pebbles are released into the top of the core one by one with the reactor operating. They are correspondingly removed from the bottom, broken ones are separated, the burn-up is measured, and spent fuel elements are screened out and transferred to storage. A 40-year operating lifetime is expected.

China Huaneng Group, one of China's major generators, is the lead organization involved in the demonstration unit with 47.5% share; China Nuclear Engineering & Construction (CNEC) has a 32.5% stake and Tsinghua University's INET 20% – it being the main R&D contributor. Projected cost is $430 million (but later units falling to $1500/kW with generating cost about 5 ¢/kWh). The HTR-PM rationale is both eventually to replace conventional reactor technology for power, and also to provide for future hydrogen production. INET is in charge of R&D, and was aiming to increase the size of the 250 MWt module and also utilize thorium in the fuel.

The 210 MWe Shidaowan HTR-PM demonstration plant at Rongcheng in Shandong province is expected to start up late in 2021, having started construction at the end of 2012. It is to pave the way for commercial 600 MWe reactor units (6x250 MWt, total 655 MWe) with a single heat exchanger and turbine, also using the steam cycle at 43.7% thermal efficiency. Plant operating lifetime is envisaged as 40 years with 85% load factor. The capital cost per kW is expected to be 75% of the small HTR-PM, and for subsequent units, 50%. Meanwhile CNEC is promoting the technology for HTR-PM 600 plants using six 250 MWt modules. Eventually a series of HTRs, possibly with Brayton cycle directly driving the gas turbines, would be factory-built and widely installed throughout China.

Performance of both this and South Africa's PBMR design includes great flexibility in loads (40-100%) without loss of thermal efficiency, and with rapid change in power settings. Power density in the core is about one-tenth of that in a light water reactor, and if coolant circulation ceases the fuel will survive initial high temperatures while the reactor shuts itself down – giving inherent safety. Power control is by varying the coolant pressure, and hence flow. (See also section on Shidaowan HTR-PM in the information page on Nuclear Power in China and the Research and development section in the information page on China's Nuclear Fuel Cycle).

Urenco U-Battery

Urenco with others commissioned a study by TU-Delft and Manchester University on the basis of which it has called for European development of a very small 'plug and play' inherently-safe reactor called the U-Battery. This is based on graphite-moderated, helium cooled HTR concepts such as the UK's Dragon reactor (to 1975). The fuel block design is based on that of the Fort St Vrain reactor in the USA. It would use TRISO fuel with 17-20% enriched uranium and possibly thorium with a beryllium oxide reflector. The 10 MWt design can produce 750°C process heat or up to 4 MWe back-up and off-grid power. The consortium envisages up to six U-Batteries at one site.

This micro-SMR U-battery would run for five years before refuelling and servicing, a larger 20 MWt one would run for 10 years. The 10 MWt/4 MWe design, 1.8 m diameter, may be capable of being returned to the factory for refuelling. The U-Battery consortium, led by Urenco, has gained UK government support for a prototype, with target operation in 2028. Wood, Laing O'Rourke, Cammell Laird and Kinectrics are involved.

In mid-2018 the consortium was one of eight organisations to be awarded a contract to produce a feasibility study as part of the UK government's Advanced Modular Reactor Feasibility and Development project, and in July 2020 it was selected for phase 2 of this. It has been accepted for pre-licensing vendor design review with the Canadian Nuclear Safety Commission (CNSC), from 2017. In July 2019 it became the first design to complete the first of the four phases of Canadian Nuclear Laboratories' review process for siting an SMR at Chalk River Laboratories in Ontario.

Russian HTR for Indonesia

In 2015 a consortium of Russian and Indonesian companies led by Nukem Technologies had won a contract for the preliminary design of the multi-purpose 10 MWe HTR in Indonesia, which would be "a flagship project in the future of Indonesia's nuclear program". It will be a pebble-bed HTR at Serpong. Atomproekt is architect general, and OKBM Afrikantov the designer. SRI Luch is also involved with fuel design. The conceptual design was completed in December 2015. In March 2018 Batan said that it aimed to complete the detailed engineering design by the end of the year, and then to call for bids to construct the reactor, for both electricity and process heat.

X-energy Xe-100

X-energy founded in 2009 in the USA is designing the Xe-100 pebble-bed HTR of 200 MWt, 80 MWe, and has been in talks with utilities, stressing that a plant will fit on a 4 ha site, below grade for electricity and/or process heat. The initial TRISO fuel in the mid-2020s will utilize uranium oxycarbide (UCO) made from high-assay low-enriched uranium (HALEU), but longer-term thorium is intended as the primary fuel. Unlike other pebble bed HTRs, the fuel will only pass through once, with high 160 GWd/t burn-up. Fairly rapid load-following from 25% to 100% is a feature of the helium-cooled design running at 750 °C. Factory-made units with 60-year operational life would be transported to the site by road and installed.

The company has been in discussion with several utilities, including South Carolina Electricity & Gas (SCEG), regarding replacing coal-fired capacity with the four-pack installations. Industrial process heat is also a likely application. X-energy is working in partnership with BWX Technology, Oregon State University, Teledyne-Brown Engineering, SGL Group, Idaho National Laboratory (INL), and Oak Ridge National Laboratory (ORNL) on the design. In January 2016 the US DOE awarded a Gateway for Accelerated Innovation in Nuclear (GAIN) grant to the project, worth $53 million. In September 2016 Burns & McDonnell Engineering joined the project as architectural and engineering partner, in parallel with the DOE five-year award. The Xe-100 is a candidate for the US Advanced Reactor Demonstration Program (ARDP). In 2020 the Xe-100 received an initial grant of $80 million under the programme.

In April 2021 X-energy signed an agreement with Energy Northwest and a public utility to set up the Tri Energy Partnership with a view to building an Xe-100 plant near the Columbia nuclear power plant in Washington state. The $2.4 billion project would be half funded by the ARDP and take seven years.

In November 2017 the company signed an agreement with Jordan Atomic Energy Commission to consider building the Xe-100 in Jordan. In August 2020 the company initiated a vendor design review with the Canadian Nuclear Safety Commission. Kinetrics is leading X-energy's Canadian regulatory affairs and licensing efforts. The company hopes to deploy the first units by 2027.

In August 2016 X-energy signed an agreement to work with Southern Nuclear Operating Company to collaborate on development and commercialization of their respective small reactor designs. Southern is developing an MSR, the Molten Chloride Fast Reactor (MCFR). In September 2018, X-energy said that its design was about 50% complete, and that it hoped the full design would be finalized by 2022 or 2023.

X-energy has a TRISO pilot fuel fabrication facility at Oak Ridge National Laboratory, Tennessee and in November 2019 it agreed with Global Nuclear Fuel (GNF) to set up commercial HALEU TRISO production at GNF's Wilmington plant. X-energy also has agreements with Centrus Energy in the USA to develop TRISO fabrication technology for uranium carbide fuel, and with NFI at Tokai in Japan, where NFI has 400 kgU/yr HTR fuel capacity.

X-energy Xe-Mobile

In March 2020 the US Department of Defense awarded a $14.3 million contract for further development of the design as a microreactor under 5 MWe – the Xe-Mobile, with all components housed in a standard shipping container. It is to be able to operate at full power – at least 1 MWe – for at least three years. In March 2021 the DOD selected this as one of two candidates to proceed to final engineering design in 2022 under the $30 million second phase of the Project Pele programme (see Military developments section above).

BWXT Advanced Nuclear Reactor

BWXT Technologies was commissioned in December 2020 by the US Department of Energy to lead a $106.6 million microreactor project under its Advanced Reactor Demonstration Program (ARDP), over seven years. It was already under a $13.5 million contract to the Department of Defense to develop a design for a transportable HTR microreactor with TRISO fuel. This is the 50 MWt BANR (BWXT Advanced Nuclear Reactor) about which few details have been released. In March 2021 the DOD selected this as one of two candidates to proceed to final engineering design in 2022 under the second phase of the Project Pele program. BWXT was awarded $28 million for this (see Military developments section above).

StarCore HTR

This is a small (20 MWe) concept design of helium-cooled reactor from StarCore Nuclear in Quebec, designed for remote locations (displacing diesel and propane) and with remote control system via satellite. It is expandable to 100 MWe. The units would be installed below grade and in pairs. They are truck-transportable, with reactor vessels 2.5 m diameter and 6 m high. Fuel is TRISO in carbon prismatic matrix. Each reactor has a five-year refuelling schedule. The secondary cooling circuit is nitrogen, to a steam generator driving a turbine. The company offers a build-own-operate-decommission concept with a power purchase agreement for the life of the reactor, mentioning C$0.18 per kWh. The units are designed to deliver both electricity and potable water.

The company has applied to the CNSC to start the pre-licensing vendor design review process.

In April 2018, Canadian Nuclear Laboratories (CNL) launched its SMR review – a separate process to licensing – with a view to having an SMR constructed on its Chalk River site by 2026. In February 2019 CNL announced that StarCore had completed the prequalification stage and been invited to enter the due diligence stage.

USNC Micro Modular Reactor

Ultra Safe Nuclear Corporation (USNC), an American company with subsidiaries in Canada and elsewhere, has the Micro Modular Reactor (MMR) HTR with the TRISO fuel in pellets in prismatic graphite blocks in a sealed transportable core. Two versions operate at 15 MWt/5 MWe or 30 MWt/10 MWe with flexible output and they require no refuelling in a 20-year operating lifetime, after which the module becomes waste. Heat is transferred from the core by helium to a molten salt system. Larger versions are envisaged.

Phase 1 of a pre-licensing vendor design review by the Canadian Nuclear Safety Commission (CNSC) was completed in February 2019, and Global First Power (GFP, jointly owned by USNC and Ontario Power Generation, OPG) then submitted a site preparation licence application for Chalk River. CNSC's environmental assessment began in July 2019. GFP, based in Ottawa, describes itself as an energy provider specializing in project development, licensing, ownership and operation of small nuclear power plants to supply clean power and heat to remote industrial operations and residential settlements. Formal licence review by the CNSC for the 15 MWt MMR began in May 2021.

In June 2020 a joint venture was formed between USNC and OPG to build, own and operate the proposed MMR project at Chalk River, Ontario. The joint venture – the Global First Power Limited Partnership – is owned equally by OPG and USNC-Power, the Canadian subsidiary of USNC. GFP said it would "provide project development, licensing, construction and operation" services for the project. The MMR would provide 15 MWt of process heat via molten salt, and have an operating lifetime of 20 years.

In August 2020 USNC signed an agreement with Hyundai Engineering and Korea Atomic Energy Research Institute for development and deployment of HTR technology for supplying power as well as process heat.

In November 2020 USNC signed an agreement with Poland's Synthos and applied to the Polish government for financing industrial-scale hydrogen projects.

In June 2021 the University of Illinois announced plans to install a USNC MMR as both a power source and research reactor at its Urbana-Champaign campus.

In April 2018, Canadian Nuclear Laboratories (CNL) launched its SMR review – a separate process to licensing – with a view to having an SMR constructed on its Chalk River site by 2026. GFP/OPG/USNC completed the first and second stages of CNL's process, and was invited to participate in the third and penultimate stage. Construction of the first 5 MWe demonstration reactor at Chalk River is expected to start in 2023, for 2025 commissioning. This will be followed by one at Idaho National Laboratory and one at the University of Illinois.

In 2020 USNC proposed an integrated solar, wind and nuclear plant providing 120 MWe of generation and 1 TWh per year for a remote defence base using ten 10 MWe MMR units. Projected power cost is 10 ¢/kWh.

(USNC is also developing an accident-tolerant shutdown system for NASA in nuclear thermal propulsion systems.)

Holos-Quad HTR

HolosGen is designing a 22 MWt micro-modular HTR in collaboration with the US military, to fit into a ISO standard 40 ft (12.2 m) shipping container. It is essentially a closed-loop jet engine (Brayton cycle) with the combustor replaced by a nuclear heat source comprising four subcritical power modules (SPMs) that are actively positioned in relation to one another, eliminating control rod mechanisms and enabling rapid load following from 3 MWe to 13 MWe. Placing the SPMs close together allows sufficient neutron transfer to reach criticality.

It uses 15% enriched TRISO fuel in graphite hexagonal blocks with 6 mm helium channels and core outlet temperature of 650-850 °C. Burnable poison is in the graphite blocks, not the fuel. Heat exchangers are embedded with the compressor components to recover waste heat for an independent organic Rankine cycle. The turbo-machinery is magnetically levitated to eliminate mechanical couplings and bearings in the core. When set up, the plant is shielded by a prefabricated structure.

Core lifetime relates to mass, and a 15-tonne core can operate for about 3.5 years, while a 27 t one can run for over eight years.

In June 2018, the HolosGen transportable reactor project was awarded $2.3 million by the Advanced Research Projects Agency-Energy (ARPA-E) of the US Department of Energy (DOE) to demonstrate the viability of the concept. An October 2018 study commissioned by the US Army put the estimated cost of a first-of-a-kind 13 MWe unit at $140 million, reducing to $75 million for later units.

HolosGen is working with Argonne National Laboratory.

Hybrid SMR concept

The hybrid-nuclear Small Modular Reactor (SMR) design from Hybrid Power Technologies LLC produces massive quantities of compressed air, while the gas turbine, able to burn a variety of fossil fuels, generates electrical power. Helium from the 600 MWt graphite-moderated reactor drives a primary turbine coupled to an air compressor. The very high pressure air then supercharges a combined cycle gas turbine (CCGT) driving an 850 MWe generator at 85% efficiency. The reactor and compressor are in a full containment structure. (The actual HTR is equivalent to less then 300 MWe output, so that component is still 'small'.) The company applied for the second round of DOE funding in 2013.

Supercritical CO_2 direct cycle fast reactor concept

This is a Generation IV design based partly on the well-proven UK advanced gas-cooled reactors (AGRs). The supercritical direct cycle gas fast reactor (SC-GFR) uses the supercritical CO_2 coolant at 20 MPa and 650°C from a fast reactor of 200 to 400 MW thermal in Brayton cycle. A small long-life reactor core could maintain decay heat removal by natural circulation. A 2011 paper from Sandia Laboratories describes it. (S-CO2 is applicable to many different heat sources, including concentrated solar. It claims high efficiency with smaller and simpler power plants. With a helium-cooled HTR or sodium-cooled fast reactor, it would be the secondary circuit.)

Antares – SC-HTGR

Another full-size HTR design is being put forward by Framatome (formerly Areva). It is based on the GT-MHR and has also involved Fuji. The reference design is 625 MWt with prismatic block fuel like the GT-MHR. Core outlet temperature is 750°C for the steam-cycle HTR version (SC-HTGR), though an eventual very high temperature reactor (VHTR) version is envisaged with 1000°C and direct cycle. The present concept uses an indirect cycle, with steam in the secondary system, or possibly a helium-nitrogen mix for the VHTR, removing the possibility of contaminating the generation, chemical or hydrogen production plant with radionuclides from the reactor core. It was selected in 2012 for the US Next Generation Nuclear Plant, with two-loop secondary steam cycle, the 625 MWt probably giving 285 MWe per unit, but the primary focus being the 750°C helium outlet temperature for industrial application. It remains at the conceptual design stage.

Adams Engine

A small HTR concept is the Adams Atomic Engines' 10 MWe direct simple Brayton cycle plant with low-pressure nitrogen as the reactor coolant and working fluid, and graphite moderation. The reactor core is a fixed, annular bed with about 80,000 fuel elements each 6 cm diameter and containing approximately 9 grams of heavy metal as TRISO particles, with expected average burn-up of 80 GWd/t. The initial units would provide a reactor core outlet temperature of 800°C and a thermal efficiency near 25%. Power output is controlled by limiting coolant flow. A demonstration plant was proposed for completion after 2018, but the design is shelved. The Adams Engine is designed to be competitive with combustion gas turbines.

An antecedent was the ML-1 nitrogen-cooled reactor with closed cycle gas turbine, designed to be air-portable and part of the US Army Nuclear Power Program (ANPP). It was water-moderated, with high-enriched fuel and from 1961 worked for several hundred hours up to two-thirds of its designed 300 kW, but various problems caused the project to be shut down in 1965. The high-pressure gas cycle with nitrogen at 910 kPa was one problem.

PBMR and derivatives

South Africa's pebble bed modular reactor (PBMR) was being developed by the PBMR (Pty) Ltd consortium led by the utility Eskom, latterly with involvement of Mitsubishi Heavy Industries, and drew on German expertise, notably the HTR-Modul design. It aimed for a step change in safety, economics and proliferation resistance. Full-scale production units had been planned to be 400 MWt (165 MWe) but more recent plans were for 200 MWt (80 MWe)[7]. Financial constraints led to delays[8] and in September 2010 the South African government confirmed it would stop funding the project[9] and closed it down.

The earlier plans for the 400 MWt PBMR following a 2002 review envisaged a direct cycle (Brayton cycle) gas turbine generator and thermal efficiency about 41%, the helium coolant leaving the bottom of the core at about 900°C and driving a turbine. Power would be adjusted by changing the pressure in the system. The helium is passed through a water-cooled pre-cooler and intercooler before being returned to the reactor vessel. The PBMR Demonstration Power Plant (DPP) was expected to start construction at Koeberg in 2009 and achieve criticality in 2013, but after this was delayed it was decided to focus on the 200 MWt design6.

The 200 MWt (80 MWe) later design announced in 2009 was to use a conventional Rankine cycle, enabling the PBMR to deliver super-heated steam via a steam generator as well as generate electricity. This design "is aimed at steam process heat applications operating at 720°C, which provides the basis for penetrating the nuclear heat market as a viable alternative for carbon-burning, high-emission heat sources."[10] An agreement with Mitsubishi Heavy Industries to take forward the R&D on this design was signed in February 2010. MHI had been involved in the project since 2001, having done the basic design and R&D of the helium-driven turbogenerator system and core barrel assembly, the major components of the 400 MWt direct-cycle design.

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

A pebble fuel plant at Pelindaba was planned. Meanwhile, the company produced some fuel which was successfully tested in Russia.

The PBMR was proposed for the US Next Generation Nuclear Plant project and submission of an application for design certification reached the pre-application review stage, but is now listed as 'inactive' by the NRC. The company was part of the National Project Management Corporation (NPMC) consortium which applied for the second round of DOE funding in 2013. This 2013 application for federal funds appeared to revive the earlier direct-cycle PBMR design, emphasising its ‘deep burn’ attributes in destroying actinides and achieving high burn-up at high temperatures.

In 2016 Eskom revived consideration of a reactor based on the PBMR, with a view to developing a design that is simpler and more efficient than the original, and also looking at applications for process heat that were not fully explored by the original R&D programme. However, most of the scientific and engineering staff had emigrated, many of them to the USA and many joined X-energy’s similar project.

A new concept was for an advanced high-temperature reactor of 150 MWe to be deployed in the 2030s, with a 50 MWe pilot plant built in the mid-2020s. It would be a combined-cycle plant with gas flow now from bottom to top, and the temperature will be much higher. The pressure vessel would be concrete, and it would have a pebble bed reactor core. Helium would exit the reactor to a gas turbine at 1200°C, and the exhaust gas from this at 600°C would drive a steam cycle, using a molten salt circuit, with overall 60% thermal efficiency. The gas turbine would produce 40% of the power, the steam cycle 60%.

A further conceptual design is the HTMR-100, a 35 MWe (100 MWt) pebble bed HTR for electricity or process heat. The conceptual design, commenced in 2012, from Steenkampskraal Thorium Limited (STL) in South Africa, was completed in 2018. Also known as the Th-100, it is derived from the Jülich and PBMR designs. For electricity, single units have load-following capability, or four can comprise a 140 MWe power plant. There are a range of fuel options involving LEU, thorium and reactor-grade plutonium, with burn-up of 80-90 GWd/t of TRISO fuel pebbles. It has a graphite moderator and helium coolant at 750°C, and a single pass fuel cycle. The reactor vessel is 15 m high, 5.9 m diameter and primary loop pressure is relatively low at 4 MPa.

GT-MHR

In the 1970s General Atomics developed an HTR with prismatic fuel blocks based on those in the 842 MWt Fort St Vrain reactor, which ran 1976-89 in the USA. Licensing review by the NRC was underway until the projects were cancelled in the late 1970s.

Evolved from this in the 1980s, General Atomics' Gas Turbine – Modular Helium Reactor (GT-MHR), would be built as modules of up to 600 MWt, but typically 350 MWt, 150 MWe. In its electrical application each would directly drive a gas turbine at 47% thermal efficiency. It could also be used for hydrogen production (100,000 t/yr claimed) and other high temperature process heat applications. The annular core, allowing passive decay heat removal, consists of 102 hexagonal fuel element columns of graphite blocks with channels for helium coolant and control rods. Graphite reflector blocks are both inside and around the core. Half the core is replaced every 18 months. Enrichment is about 15.5%, burn-up is up to 220 GWd/t, and coolant outlet temperature is 750°C with a target of 1000°C.

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

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

A smaller version of the GT-MHR, the Remote-Site Modular Helium Reactor (RS-MHR) of 10-25 MWe was proposed by General Atomics. The fuel would be 20% enriched and the refuelling interval would be 6-8 years.

EM^2

In February 2010, General Atomics announced its Energy Multiplier Module (EM^2) fast neutron design, superseding its GT-MHR. The EM^2 is a 500 MWt, 265 MWe helium-cooled fast-neutron HTR operating at 850°C to achieve 53% net thermal efficiency with a variety of fuels and using the Brayton cycle. It has several passive safety features and in particular the fuel rod cladding is manufactured from GA's proprietary SiGA silicon-carbide composite, a high-tech ceramic matrix composite that can withstand more than twice the temperatures of the metal components used in most reactors. Decay heat removal is entirely passive.

The EM^2 may be fuelled with 20 tonnes of used PWR fuel or depleted uranium, plus 22 tonnes of low-enriched uranium (~12% U-235, HALEU) as starter. Used fuel from this is processed to remove fission products (about 4 tonnes) and the balance is recycled as fuel for subsequent rounds, each time topped up with 4 tonnes of further used PWR fuel. Each refuelling cycle may be as long as 30 years. With repeated recycling the amount of original natural uranium (before use by PWR) used goes up from 0.5% to 50% at about cycle 12. High-level waste is about 4% of that from PWR on open fuel cycle. EM^2 would also be suitable for process heat applications. The main pressure vessel can be trucked or railed to the site, and installed below ground level, and the high-speed (gas) turbine generator is also truck-transportable. The company expects a four-unit EM^2 plant to be built in 42 months. The means of reprocessing to remove fission products is not specified, except that it is not a wet process. The company applied for the second round of DOE funding in 2013.

The company anticipates a 12-year development and licensing period, which is in line with the 80 MWt experimental technology demonstration gas-cooled fast reactor (GFR) in the Generation IV programme[l]. GA has teamed up with Chicago Bridge & Iron, Mitsubishi Heavy Industries, and Idaho National Laboratory to develop the EM^2.

GA-Framatome Fast Modular Reactor

General Atomics Electromagnetic Systems Group (GA-EMS) in the USA is collaborating with Framatome Inc. (the US branch of Framatome) to develop a new helium-cooled 50 MWe design, the Fast Modular Reactor (FMR), primarily for electricity using the Brayton cycle at 45% thermal efficiency. The refuelling cycle would be nine years, apparently using GA's proprietary SiGA silicon-carbide composite fuel cladding, though no information about fuel has been announced. It will be dry-cooled regarding waste heat, with passive safety. It will have fast-response load-following capability of about 20% per minute ramping while maintaining reactor temperature to mitigate thermal cycle fatigue in components. It will be factory-built and assembled onsite. Framatome's US engineering team will be responsible for designing several critical structures, systems and components for the FMR. A demonstration unit is expected to operate in early 2030s. Operating temperature is expected to be over 700 °C (cf 850 °C for EM^2 at higher thermal efficiency)

GA-EMS is separate from General Atomics' Energy Group, which is developing the Energy Multiplier Module (EM^2). GE-EMS is best known for the electromagnetic aircraft launch and recovery systems fitted to the latest US aircraft carriers, as well as rail guns and hypervelocity projectiles.

Fast neutron reactors

Fast neutron reactors (FNR) are smaller and simpler than light water types, they have better fuel performance and can have a longer refueling interval (up to 20 years), but a new safety case needs to be made for them, at least in the west. They are designed to use the full energy potential of uranium, rather than about one percent of it that conventional power reactors use. They have no moderator, a higher neutron flux and are normally cooled by liquid metal such as sodium, lead, or lead-bismuth, with high conductivity and boiling point. They operate at or near atmospheric pressure and have passive safety features (most have convection circulating the primary coolant). Automatic power regulation is achieved due to the reactivity feedback – loss of coolant flow leads to higher core temperature which slows the reaction. Fast reactors typically use boron carbide control rods.

Fuels are mostly 15-20% enriched and may be uranium nitride – UN, (U,Pu)N, (U,transuranic)N, or (U,Pu)Zr. In the USA no enrichment plant is designed for more than 10% enrichment, but the government has 26 tonnes of HEU unallocated, and this might be blended down for fast reactors.

Most coolants are liquid metal, either sodium, which is flammable and reacts violently with water, or lead/lead-bismuth, which is corrosive but does not react with air or water. It eliminates the need and associated expense of extra components and redundant safety systems required by other technologies for protection against coolant leakages. Both coolants can be used at or near atmospheric pressure, which simplifies engineering and reduces cost. Their high-temperature operation benefits thermodynamic efficiency.

There are two exceptions to liquid metal cooling: gas and salt.

Two gas-cooled fast reactor (GFR) concepts – the Energy Multiplier Module (EM^2) and Fast Modular Reactor (FMR) – have been announced by General Atomics and are described in the HTR section above. The concept is also being pursued in the Generation IV programme, with Allegro (50-100 MWt) being developed by the V4G4 Centre in Eastern Europe with French support. In May 2021 the Czech nuclear research institute, UJV Rez, announced its Hefasto project based on Allegro, to develop a 200 MWt reactor operating at up to 900°C. Three versions will be pitched to heating, cogeneration and the chemical industry.

Salt cooling is in the molten chloride fast reactor (MCFR) concept being developed by Southern Company Services in the USA with TerraPower, Oak Ridge National Laboratory (ORNL) and EPRI. The pilot version of this will be built at Idaho National Laboratory. Also the lead version of the Moltex stable salt reactor is fast. These are described in the Molten salt reactors section below.

Small FNRs are designed to be factory-built and shipped to site on truck, train or barge and then shipped back again or to a regional fuel cycle centre at end of life. They would mostly be installed below ground level and with high surface area to volume ratio they have good passive cooling potential. Disposal is envisaged as entire units, without separate spent fuel storage, or after fuel removed for reprocessing.

See also Fast Neutron Reactors paper.

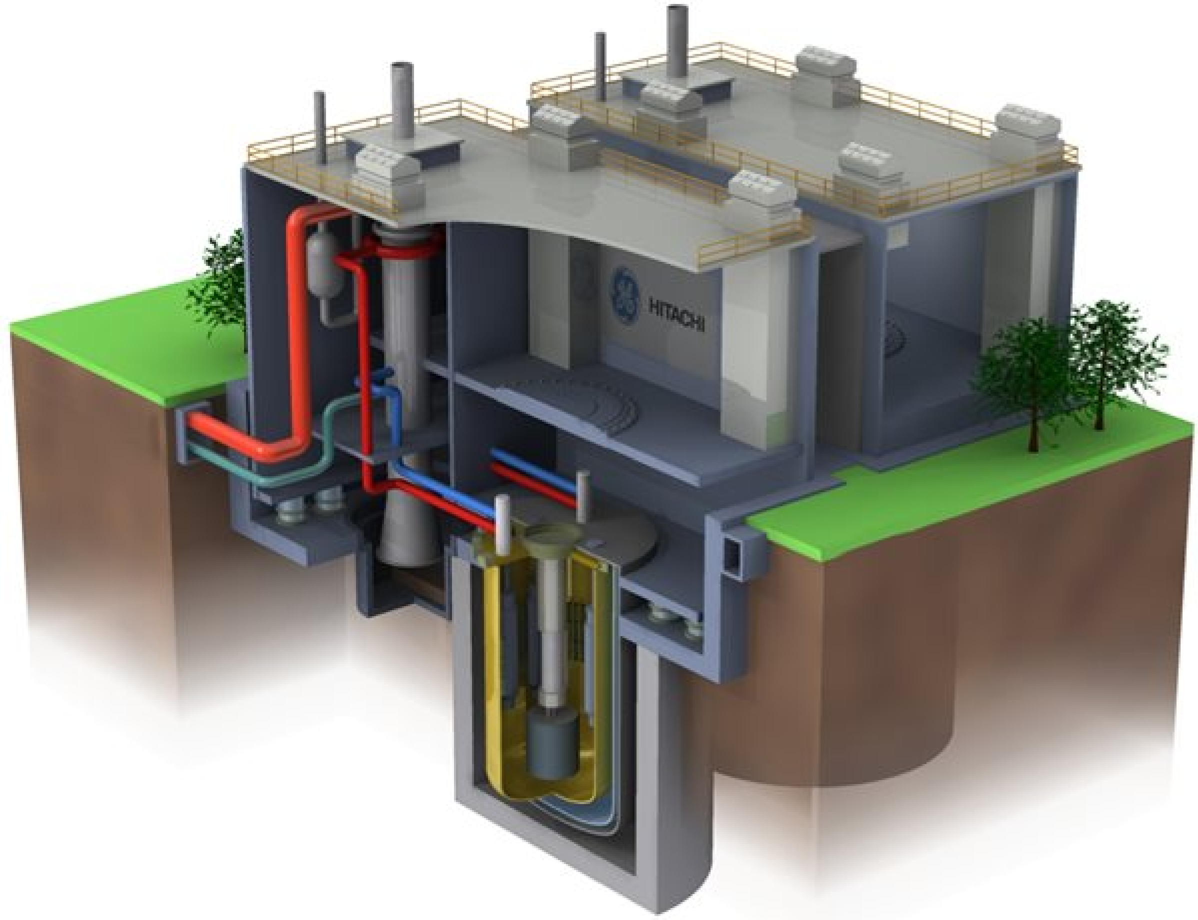
Sodium-cooled fast reactors

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

PRISM, Natrium

GE with the US national laboratories had been developing a modular liquid metal-cooled inherently-safe reactor – PRISM (Power Reactor Innovative Small Module) – under the Advanced Liquid Metal Reactor/Integral Fast Reactor (ALMR/IFR) program funded by the US Department of Energy. The design is based on EBR-II and the original IFR. Another antecedent was GE's fast reactor power plant for USS Seawolf 1957-58. The ALMR/IFR program was cancelled in 1994 and no US fast neutron reactor has so far been larger than 66 MWe and none has supplied electricity commercially. However, the 1994 pre-application safety evaluation report[13] for the original PRISM design concluded that "no obvious impediments to licensing the PRISM design had been identified."

Today's PRISM is a GE Hitachi (GEH) design for compact modular pool-type reactors with passive cooling for decay heat removal. After 30 years of development it represents GEH's Generation IV solution to closing the fuel cycle in the USA. Each PRISM power block consists of two modules of 311 MWe (840 MWt) each, (or, earlier, three modules of 155 MWe, 471 MWt), each with one steam generator, that collectively drive one turbine generator. The pool-type modules below ground level contain the complete primary system with sodium coolant at about 500°C. An intermediate sodium loop takes heat to steam generators. The metal Pu & DU fuel is obtained from used light water reactor fuel. All transuranic elements are removed together in the electrometallurgical reprocessing so that fresh fuel has minor actinides with the plutonium and uranium.



A cutaway of the PRISM design (GE Hitachi)

The reactor is designed to use a heterogeneous metal alloy core with 192 fuel assemblies in two fuel zones. In the version designed for used LWR fuel recycle, all these are fuel, giving peak burnup of 122 GWd/t. In other versions for breeding or weapons plutonium consumption, 42 of them are internal blanket and 42 are radial blanket, with 108 as driver fuel, and peak burnup of 144 GWd/t. For the LWR fuel recycle version, fuel stays in the reactor four years, with one-quarter removed annually, and 72 kg/yr net of fissile plutonium consumed. In the breeder version fuel stays in the reactor about six years, with one-third removed every two years, and net production of 57 kg/yr of fissile plutonium. 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, though not necessarily into PRISM units.

The commercial-scale plant concept, part of an 'Advanced Recycling Center', would use three power blocks (six reactor modules) to provide 1866 MWe. In 2011 GE Hitachi announced that it was shifting its marketing strategy to pitch the reactor directly to utilities as a way to recycle excess plutonium while producing electricity for the grid. GEH bills it as a simplified design with passive safety features and using modular construction techniques. Its reference construction schedule is 36 months. In October 2016 GEH signed an agreement with Southern Nuclear Development, a subsidiary of Southern Nuclear Operating Company, to collaborate on licensing fast reactors including PRISM. In June 2017 GEH joined a team led by High Bridge Energy Development Co. and including Exelon Generation, High Bridge Associates and URS Nuclear to license PRISM.

GEH is promoting to UK government agencies the potential use of PRISM technology to dispose of the UK's plutonium stockpile. Two PRISM units would irradiate fuel made from this plutonium (20% Pu, with DU and zirconium) for 45-90 days, bringing it to 'spent fuel standard' of radioactivity, after which it would be stored in air-cooled silos. The whole stockpile could be irradiated thus in five years, with some by-product electricity (but frequent interruptions for fuel changing) and the plant would then proceed to re-use it for about 55 years solely for 600 MWe of electricity generation, with one-third of the fuel being changed every two years. For this UK version, the breeding ratio is 0.8. No reprocessing plant ('Advanced Recycling Center') is envisaged initially, but this could be added later.

In March 2017 GEH and Advanced Reactor Concepts (see below) signed an agreement to collaborate on licensing an SMR design based on the ARC-100, but drawing on the extensive intellectual property and licensing experience of the GEH PRISM programme. Initial deployment is envisaged in Canada, at Point Lepreau in New Brunswick. ARC will seek a preliminary regulatory review with the CNSC through its Vendor Design Review process.

In February 2019 the US DOE launched its Versatile Test Reactor (VTR) programme, set up under the Nuclear Energy Innovation Capabilities Act 2017 and run by Idaho National Laboratory. The programme aims to provide the capability for testing advanced nuclear fuels, materials, instrumentation, and sensors. The VTR, which is intended to be operational at INL by the end of 2025, would be an adapted PRISM reactor to provide accelerated neutron damage rates 20 times greater than current water-cooled test reactors. (The only other fast research reactor operating is the BN-60 in Russia, to be replaced after 2020 by MBIR there.) In January 2020 GEH and TerraPower announced a collaboration to pursue a public-private partnership to design and construct the VTR for the DOE. They would be supported by the Energy Northwest utility consortium.

A further collaboration between GE Hitachi and Terrapower is the Natrium concept. This is based on a PRISM reactor of 345 MWe and uses molten salt to store heat so that the output could be increased to about 500 MWe for up to five hours for load-following. The primary coolant is sodium, the secondary coolant is molten salt which can store heat or use it to make steam in a heat exchanger, switching between the two as required so that plant output can vary between 30% and 150% of reactor power. It would "help customers capitalize on peaking opportunities driven by renewable energy fluctuations." Natrium is part of the DOE Advanced Reactor Demonstration Program (ARDP) offering funds on a cost-share basis and in October 2020 was awarded an initial grant of $80 million. In October 2020 Bechtel joined the consortium to provide design, licensing, procurement and construction services to the project.

In June 2021 TerraPower announced plans to build a demonstration Natrium unit in Wyoming at a retired coal plant site. It plans to submit a construction permit application in 2023 and an operating licence application in 2026. The plant is expected to cost under $1 billion apart from financing.

See also *Electrometallurgical 'pyroprocessing'* section in the information page on Processing of Used Nuclear Fuel.

Integral Fast Reactor, ARC-100

Advanced Reactor Concepts LLC (ARC) set up in 2006 has developed a 260 MWt/100 MWe sodium-cooled fast reactor based on the 62.5 MWt Experimental Breeder Reactor II (EBR-II). It will be factory-produced, with components readily assembled onsite, and with 'walk-away' passive safety. Installation would be below ground level.

The ARC-100 system comprises a uranium alloy metal core cartridge submerged in sodium at ambient pressure in a stainless steel tank. The liquid sodium is pumped through the core where it is heated to 510°C, then passed through an integral heat exchanger (within the pool) where it heats sodium in an intermediate loop, which in turn heats working fluid for electricity generation. It would have a refuelling interval of 20 years for cartridge changeover, with 20.7 tonnes of fuel. Initial fuel will be low-enriched uranium (10.1% inner zone, 12.1% middle zone, 17.2% outer zone among 92 fuel assemblies over 1.5 m fuelled height) but it will be able to burn wastes from light water reactors, or plutonium. Reprocessing its used fuel will not separate plutonium. ARC-100 has load-following capability. Thermal efficiency is about 40% and it and could be paired with a supercritical carbon dioxide tertiary circuit to drive a turbine at high efficiency. Operating cost is expected to be $50/MWh.

In March 2017 GEH and ARC signed an agreement to collaborate on licensing an SMR design based on ARC-100, which will leverage extensive intellectual property and licensing experience of the GEH PRISM programme. A further agreement in August 2017 licensed PRISM technology to ARC, and provided GEH engineering and design expertise to ARC. Initial deployment is envisaged in Canada by ARC Canada, and in October 2019 the CNSC completed phase 1 pre-licensing vendor design review for the ARC-100.

In July 2018 ARC and New Brunswick Power announced that they were exploring the potential deployment of the ARC-100 reactor at New Brunswick's Point Lepreau nuclear plant, and in November 2020 the two companies were joined by Moltex in setting up an SMR vendor cluster there. In February 2021 the New Brunswick government announced $20 million funding for ARC Canada and in April 2021 plans for the first unit at Point Lepreau were confirmed. In 2021 ARC offered the design to Energoatom in Ukraine.

CEFR

The China Experimental Fast Reactor of 65 MWt is basically that, rather than a power reactor, though it can incidentally generate 20 MWe. It is an important part of China's reactor development, and details are in the R&D section of the China Fuel Cycle paper. It is sodium-cooled at 530°C and has been operating since 2010.

Rapid-L

A small-scale design developed by Japan's Central Research Institute of Electric Power Industry (CRIEPI) in cooperation with Mitsubishi Research Institute and funded by the Japan Atomic Energy Research Institute (JAERI) is the 5 MWt, 200 kWe Rapid-L, using lithium-6 (a neutron poison) as control medium. It would have 2700 fuel pins of 40-50% enriched uranium nitride with

2600°C melting point integrated into a disposable cartridge or 'integrated fuel assembly'. The reactivity control system is passive, using lithium expansion modules (LEMs) which give burn-up compensation, partial load operation as well as negative reactivity feedback. During normal operation, lithium-6 in the LEM is suspended on an inert gas above the core region. As the reactor temperature rises, the lithium-6 expands, moving the gas/liquid interface down into the core and hence adding negative reactivity. Other kinds of lithium modules, also integrated into the fuel cartridge, shut down and start up the reactor. Cooling is by molten sodium, and with the LEM control system, reactor power is proportional to primary coolant flow rate. Refuelling would be every 10 years in an inert gas environment. Operation would require no skill, due to the inherent safety design features. The whole plant would be about 6.5 metres high and 2 metres diameter.

The larger RAPID reactor delivers 1 MWe and is U-Pu-Zr fuelled and sodium-cooled.

4S

The Super-Safe, Small & Simple (4S) 'nuclear battery' system is being developed by Toshiba and the Central Research Institute of Electric Power Industry (CRIEPI) in Japan in collaboration with SSTAR work and Westinghouse (owned by Toshiba) in the USA. It uses sodium as coolant (with electromagnetic pumps) and has passive safety features, notably negative temperature coefficient of reactivity. The whole unit would be factory-built, transported to site, installed below ground level, and would drive a steam cycle via a secondary sodium loop. It is capable of three decades of continuous operation without refuelling. Metallic fuel (169 pins 10mm diameter) is uranium-zirconium enriched to less than 20% or U-Pu-Zr alloy with 24% Pu for the 30 MWt (10 MWe) version or 11.5% Pu for the 135 MWt (50 MWe) version. Steady power output over the core lifetime in 30 MWt version is achieved by progressively moving upwards an annular reflector around the slender core (0.68m diameter, 2m high in the small version; 1.2m diameter and 2.5m high in the larger version) at about one millimetre per week. After 14 years a neutron absorber at the centre of the core is removed and the reflector repeats its slow movement up the core for 16 more years. Burn-up will be 34 GWday/t. In the event of power loss the reflector falls to the bottom of the reactor vessel, slowing the reaction, and external air circulation gives decay heat removal. A further safety device is a neutron absorber rod which can drop into the core. After 30 years the fuel would be allowed to cool for a year, then it would be removed and shipped for storage or disposal.

Both versions of 4S are designed to automatically maintain an outlet coolant temperature of 510-550ºC – suitable for power generation with high temperature electrolytic hydrogen production. Plant cost is projected at US$ 2500/kW and power cost 5-7 cents/kWh for the small unit – very competitive with diesel in many locations. The design has gained considerable support in Alaska and toward the end of 2004 the town of Galena granted initial approval for Toshiba to build a 10 MWe (30 MWt) 4S reactor in that remote location. A pre-application Nuclear Regulatory Commission (NRC) review was under way to 2008 with a view to application for design certification in October 2010, and combined construction and operating licence (COL) application to follow. Its review is now listed as 'inactive' by NRC. Its design is sufficiently similar to PRISM – GE's modular 150 MWe liquid metal-cooled inherently-safe reactor which went part-way through the NRC approval process (see section above on PRISM) – for it to have good prospects of licensing. Toshiba planned a worldwide marketing program to sell the units for power generation at remote mines, for extraction of tar sands, desalination plants and for making hydrogen. Eventually it expected sales for hydrogen production to outnumber those for power supply.

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

Travelling wave and standing wave reactors

This is not a small reactor, and details are in the information page on Fast Neutron Reactors and at TerraPower.

Lead- and lead-bismuth cooled fast reactors

Lead or lead-bismuth eutectic in fast neutron reactors are capable of high temperature operation at atmospheric pressure. Pb-208 – 54% of naturally-occurring lead – is transparent to neutrons. This means that efficiency is better due to greater spacing between fuel pins which then allows coolant flow by convection for decay heat removal. Also since they do not react with water the heat exchanger interface is safer. They do not burn when exposed to air. However, they are corrosive of fuel cladding and steels, which originally limited temperatures to 550°C. With today's materials 650°C can be reached, and in future 800°C is envisaged with the second stage of Generation IV development, using oxide dispersion-strengthened steels. Lead and Pb-Bi have much higher thermal conductivity than water, but lower than sodium.

While lead has limited activation from neutrons, a problem with Pb-Bi is that it yields toxic polonium (Po-210) activation product, an alpha-emitter with a half-life of 138 days. Pb-Bi melts at a relatively low 125°C (hence eutectic) and boils at 1670°C, Pb melts at 327°C and boils at 1737°C but is very much more abundant and cheaper to produce than bismuth, hence is envisaged for large-scale use in the future, though freezing must be prevented. In 1998 Russia declassified a lot of research information derived from its experience with Pb-Bi in submarine reactors, and US interest in using Pb generally or Pb-Bi for small reactors has increased subsequently.

BREST-300

Russia has experimented with several lead-cooled reactor designs, and gained 70 reactor-years experience with lead-bismuth cooling to 1990s in submarine reactors. A significant new Russian design from NIKIET is the BREST fast neutron reactor, of 700 MWt, 300 MWe, with lead as the primary coolant, at 540°C, supplying supercritical steam generators. The core sits in a pool of lead at near atmospheric pressure. It is inherently safe and uses a U+Pu nitride fuel. Effective enrichment is about 13.5%. Fuel cycle is quoted at 5-6 years with partial refuelling at about 10 months. No weapons-grade plutonium can be produced (since there is no uranium blanket), and used fuel can be recycled indefinitely, with on-site facilities.

The pilot demonstration unit is being built at Seversk for completion in 2026, and 1200 MWe units are planned. The BREST reactor is an integral part of the Pilot Demonstration Energy Complex (PDEC) which comprises three elements: a mixed uranium-plutonium nitride fuel fabrication/re-fabrication module; a nuclear power plant with BREST-300 reactor; and a used nuclear fuel reprocessing module (for 2024 operation). The combination enables a fully closed fuel cycle on one site.

SVBR-100

A smaller and newer Russian design as a small modular reactor was to be the lead-bismuth fast reactor (SVBR) of 280 MWt, 100 MWe, being developed by AKME-engineering and involving Gidropress in the design. It is an integral design, with 12 steam generators and two main circulation pumps sitting in the same Pb-Bi pool at 340-490°C as the reactor core. It is designed to be able to use a wide variety of fuels, though the pilot unit would initially use uranium oxide enriched to 16.3%. With U-Pu MOX fuel it would operate in closed cycle. Refuelling interval would be 7-8 years and 60-year operating lifetime was envisaged. The melting point of the Pb-Bi coolant is 123.5°C, so it is readily kept molten during shutdown by decay heat supplemented by external heat sources if required.

The SVBR-100 unit of 280 MWt would be factory-made and transported (railway, road or waterway) as a 4.5m diameter, 8.2m high module. A power station with such modules was expected to supply electricity at lower cost than any other new technology with an equal capacity as well as achieving inherent safety and high proliferation resistance. (Russia built seven Alfa-class submarines, each powered by a compact 155 MWt Pb-Bi cooled reactor, and 80 reactor-years' operational experience was acquired with these.) In October 2015 Rosatom reported: "Experts have confirmed there are no scientific or technical issues that would prevent completion of the project and obtaining a construction licence." Then in November 2016 Rosatom said it expected to work out the main specifications for construction of the SVBR-100 by mid-2017, but in 2018 the project was dropped. Overnight capital cost was earlier estimated as $4000-4500/kW and generating costs 4-5 c/kWh on 90% load factor.

In December 2009, AKME-engineering, a 50-50 joint venture, was set up by Rosatom and the En+ Group (a subsidiary of Basic Element Group) as an open joint stock company to develop and build a pilot SVBR unit[14]. En+ is an associate of JSC EuroSibEnergo and a 53.8% owner of Rusal, which had been in discussion with Rosatom regarding a Far East nuclear power plant and Phase II of the Balakovo nuclear plant. It was to contribute most of the capital, and Rosatom is now looking for another investor. In 2011 the EuroSibEnergo 50% share passed to its subsidiary JSC Irkutskenergo. The main project participants are OKB Gidropress at Podolsk, VNIPIET OAO at St Petersburg, and the RF State Research Centre Institute of Physics & Power Engineering (IPPE or FEI) at Obninsk.

The plan was to complete the design development and put online a 100 MWe pilot facility by 2019, with total investment of RUR36 billion ($550 million). The site was to be the Research Institute of Atomic Reactors (RIAR or NIIAR) at Dimitrovgrad – Russia's largest nuclear research centre – though earlier plans were to put it at IPPE/FEI at Obninsk. The SVBR-100 would have been the first reactor cooled by heavy metal to generate electricity. It is described by Gidropress as a multi-function reactor for power, heat or desalination.

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

(Link to SVBR brochure)

Gen4 (Hyperion) Power Module

The Gen4 Module is a 70 MWt/25 MWe lead-bismuth cooled reactor concept using 19.75% enriched uranium nitride fuel, from Gen4 Energy. The reactor was originally conceived as a potassium-cooled self-regulating 'nuclear battery' fuelled by uranium hydride[m]. However, in 2009, Hyperion Power changed the design to uranium nitride fuel and lead-bismuth cooling to expedite design certification[12]. This now classes it as a fast neutron reactor, without moderation. The company claims that the ceramic nitride fuel has superior thermal and neutronic properties compared with uranium oxide. Enrichment is 19.75% and operating temperature about 500°C. The lead-bismuth eutectic is 45% Pb, 55% Bi. The unit would be installed below ground level.

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

In March 2010, Hyperion (as the company then was) notified the US Nuclear Regulatory Commission that it planned to submit a design certification application in 2012. The company said then that it has many expressions of interest for ordering units. In September 2010, the company signed an agreement with Savannah River Nuclear Solutions to possibly build a demonstration unit at the Department of Energy site there. Hyperion planned to build a prototype by 2015, possibly with uranium oxide fuel if the nitride were not then available. In March 2012 the US DOE signed an agreement with Hyperion regarding constructing a demonstration unit at its Savannah River site in South Carolina.

In 2014 two papers on nuclear marine propulsion were published arising from a major international industry project led by Lloyd's Register. They describe a preliminary concept design study for a 155,000 dwt Suezmax tanker that is based on a conventional hull form with a 70 MW Gen4 Energy power module for propulsion.

In March 2012 Hyperion Power Generation changed its name to Gen4 Energy, and the name of its reactor to Gen4 Module (G4M). It pitched its design for remote sites having smaller power requirements.

Westinghouse LFR

The Westinghouse Lead-cooled Fast Reactor (LFR) programme originated from an investigation performed in 2015 aimed at identifying the technology that would best support addressing the challenges of nuclear power, for global deployment. It is at the conceptual design stage for up to 450 MWe as a modular pool-type unit, simple, scalable and with passive safety. It will have flexible output to complement intermittent renewable feed to the grid. Its high temperature – eventually 650°C – capabilities will allow industrial heat applications. Westinghouse expects it to be very competitive, having low capital and construction costs with enhanced safety.

Because lead coolant operates at atmospheric pressure and does not exothermically react with air or with power conversion fluids (such as supercritical carbon dioxide and water), LFR technology also eliminates the need and associated expense of extra components and redundant safety systems required by other plant designs for protection against coolant leakages. 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 programme.

In February 2017 the company signed an agreement with the Italian National Agency for New Technologies, Energy and Sustainable Economic Development (ENEA) and Ansaldo Nucleare to develop the design. The development also involves several UK companies and initial licensing is envisaged with the UK Office for Nuclear Regulation (ONR). In April 2021 an Ansaldo subsidiary was contracted to design, provide, install and test key components of the reactor at the Versatile Lead Loop Facility and Passive Heat Removal Facility, which are to be designed and installed at Ansaldo Nuclear's site in Wolverhampton in the UK. A prototype LFR will be about 300 MWe, running at 500 °C.

Beyond base-load electricity generation, the high-temperature operation of the LFR will allow for effective load-following capability enabled by an innovative thermal energy storage system, as well as delivery of process heat for industrial applications and water desalination. A supercritical carbon dioxide power conversion system that uses air as the ultimate heat sink significantly reduces water utilization and eliminates the need for siting the plant near large water bodies.

Encapsulated Nuclear Heat-Source

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

The heatpipe ENHS has the heat removed by liquid-metal heatpipes. Like the SAFE-400 space nuclear reactor core, the HP-ENHS core comprises fuel rods and heatpipes embedded in a solid structure arranged in a hexagonal lattice in a 3:1 ratio. The core is oriented horizontally and has a square rather than cylindrical cross-section for effective heat transfer. The heatpipes extend from the two axial reflectors in which the fission gas plena are embedded and transfer heat to an intermediate coolant that flows by natural circulation. (The SAFE-400 space fission reactor – Safe Affordable Fission Engine – was a 400 kWt heatpipe power system of 100 kWe to power a space vehicle using two Brayton power systems (gas turbines driven directly by the hot gas from the reactor.)

STAR-LM, STAR-H2, SSTAR

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

The STAR-H2 is an adaptation of the same reactor for hydrogen production, with reactor heat at up to 800°C being conveyed by a helium circuit to drive a separate thermochemical hydrogen production plant, while lower grade heat is harnessed for desalination (multi-stage flash process). Its development is further off.

A smaller STAR variant is the Small Sealed Transportable Autonomous Reactor (SSTAR) which was being developed by Lawrence Livermore, Argonne and Los Alamos National Laboratories in collaboration with others including Toshiba. It has lead or Pb-Bi cooling, 564°C core outlet temperature and has integral steam generator inside the sealed unit, which would be installed below ground level. Conceived in sizes 10-100 MWe, main development was focused on a 45 MWt/20 MWe version as part of the US Generation IV effort. After a 20- or 30-year operating lifetime without refuelling, the whole reactor unit is then returned for recycling the fuel. The reactor vessel is 12 metres high and 3.2 m diameter and the core one metre high and 1.2 m diameter (20 MWe version). SSTAR would eventually be coupled to a Brayton cycle turbine using supercritical carbon dioxide with natural circulation to four heat exchangers. A prototype was envisaged for 2015, but development has apparently ceased.

LSPR

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

SEALER

LeadCold Reactors (Blykalla Reaktorer) was founded in 2013 as a spin-off company from the Royal Institute of Technology (KTH) in Stockholm. It has a subsidiary in Canada. Its SEALER-3 (Swedish Advanced Lead Reactor) is a lead-cooled fast reactor designed with the smallest possible core that can achieve criticality in a fast spectrum using 20% enriched uranium oxide fuel. The basic reactor is 8 MWt, with a peak electric power of 3 MWe, leading to a core life of 30 full power years (at 90% availability with no refuelling) with coolant below 450°C to minimise corrosion. The company has developed novel aluminium-steel alloys that are highly corrosion-resistant in contact with liquid lead up to 450°C. The reactor vessel is designed to be small enough to permit transportation by aircraft.

As the regulatory framework for licensing of small reactors in Canada is better established than in most other countries, Nunavut and the Northwest Territories are likely to become the first markets for SEALER units. The Canadian Nuclear Safety Commission (CNSC) commenced phase 1 of a 15-month pre-licensing vendor design review in January 2017, but the review is now on hold at the vendor's request. In 2016 an Essel Group Middle East subsidiary agreed to invest in the Swedish-Canadian project, and in January 2017 a $200 million investment agreement was signed to license and construct "the world's first privately funded lead-cooled nuclear power plant." The funding will enable LeadCold to complete the pre-licensing review with the CNSC, complete a detailed engineering design of the reactor, carry out the R&D necessary for licensing the design in Canada, and construct a full-scale 3 MWe demonstration unit by about 2025. In April 2018 the company began collaboration on safety analysis with Netherlands-based NRG, which operates the Petten high-flux research reactor.

In February 2021 Uniper Sweden signed a joint venture agreement, creating Swedish Modular Reactors AB, with LeadCold and KTH aimed at constructing a demonstration SEALER-3 by 2030 at Oskarshamn. In February 2022 the Swedish Energy Agency awarded the joint venture funding of $10.6 million.

SEALER-5 is a 5 MWe reactor design. Replacing the standard uranium oxide fuel with uranium nitride (UN), the same core can host 40% more fissile material. This allows the core to operate at 40% higher thermal power for the same duration as SEALER-3, *i.e.* 30 years.

SEALER-10 is the waste management system. After 30 years of operation, the early SEALER units will be transported back to a centralised recycling facility. The plutonium and minor actinides present in the spent fuel will then be separated and converted into nitride fuel for recycle in a 10 MWe SEALER reactor. One such reactor will be sufficient to manage the used fuel of ten smaller SEALER units.

Chinese Hedianbao

A small research institute at Hefei, Anhui province in China is doing some conceptual work on a “portable nuclear battery pack” designed to fit inside a standard shipping container. The lead-cooled fast reactor would be able to generate 10 megawatts thermal, and is based on a Russian submarine reactor design.

Korean fast reactor designs

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

Molten salt reactors

These mostly use molten fluoride salts as primary coolant, at low pressure. Lithium-beryllium fluoride and lithium fluoride salts remain liquid without pressurization up to 1400°C, in marked contrast to a PWR which operates at about 315°C under 150 atmospheres pressure. Fast-spectrum MSRs use chloride salt coolant. In most designs the fuel is dissolved in the primary coolant, but in some the fuel is a pebble bed.

During the 1960s, the USA developed the molten salt breeder reactor concept as the primary back-up option for the fast breeder reactor (cooled by liquid metal) and a small prototype 8 MWt Molten Salt Reactor Experiment (MSRE) operated at Oak Ridge over four years to 1969 (the MSR programme ran 1957-1976). U-235 tetrafluoride enriched to 33% was in molten lithium, beryllium and zirconium fluorides at 600-650°C which flowed through a graphite moderator. A second campaign used U-233 fuel, but the program did not progress to building a MSR breeder utilising thorium. There is now renewed interest in the concept in Japan, Russia, China, France and the USA, and one of the six Generation IV designs selected for further development is the molten salt reactor (MSR).

In the normal MSR, the fuel is a molten mixture of lithium and beryllium fluoride (FLiBe) salts with dissolved enriched uranium – U-235 or U-233 fluorides (UF_4). The core consists of unclad graphite moderator arranged to allow the flow of salt at some 700°C and at low pressure. Much higher temperatures are possible but not yet tested. Heat is transferred to a secondary salt circuit and thence to steam[o]. The basic design is not a fast neutron reactor, but with some moderation by the graphite, may be epithermal (intermediate neutron speed) and breeding ratio is less than 1.

Thorium can be dissolved with the uranium in a single fluid MSR, known as a homogeneous design. Two-fluid, or heterogeneous MSRs would have fertile salt containing thorium in a second loop separate from the fuel salt containing fissile uranium and could operate as a breeder reactor (MSBR). In each case secondary coolant salt circuits are used.

The fission products dissolve in the fuel salt and may be removed continuously in an on-line reprocessing loop and replaced with fissile uranium or, potentially, Th-232 or U-238. Actinides remain in the reactor until they fission or are converted to higher actinides which do so.

The liquid fuel has a negative temperature coefficient of reactivity and a strong negative void coefficient of reactivity, giving passive safety. If the fuel temperature increases, the reactivity decreases. The MSR thus has a significant load-following capability where reduced heat abstraction through the boiler tubes leads to increased coolant temperature, or greater heat removal reduces coolant temperature and increases reactivity. Primary reactivity control is using the secondary coolant salt pump or circulation which changes the temperature of the fuel salt in the core, thus altering reactivity due to its strong negative reactivity coefficient. The MSR works at near atmospheric pressure, eliminating the risk of explosive release of volatile radioactive materials.

Other attractive features of the MSR fuel cycle include: the high-level waste comprising fission products only, hence shorter-lived radioactivity (actinides are less-readily formed from U-233 than in fuel with atomic mass greater than 235); small inventory of weapons-fissile material (Pu-242 being the dominant Pu isotope); high temperature operation giving greater thermal efficiency; high burn-up of fuel and hence low fuel use (the French self-breeding variant claims 50kg of thorium and 50kg U-238 per billion kWh); and safety due to passive cooling up to any size. Several have freeze plugs so that the primary salt can be drained by gravity into dump tanks configured to prevent criticality. Control rods are actually shut-down rods.

Lithium used in the primary salt must be fairly pure Li-7, since Li-6 produces tritium when fissioned by neutrons. Li-7 has a very small neutron cross section. This means that natural lithium must be enriched, and is costly. Pure Li-7 is not generally used in secondary coolant salts. But even with enriched Li-7, some tritium is produced and must be retained and recovered.

The MSR concept is being pursued in the Generation IV programme with two variants: one a fast neutron reactor with fissile material dissolved in the circulation fuel salt, and with solid particle fuel in graphite and the salt functioning only as coolant.

MSRs would normally operate at much higher temperatures than LWRs – up to at least 700°C, and hence have potential for process heat. Molten fluoride salts (possibly simply cryolite – Na-Al fluoride) are a preferred interface fluid in a secondary circuit between the nuclear heat source and any chemical plant. The aluminium smelting industry provides substantial experience in managing them safely.

One MSR developer, Moltex, has put forward a molten salt heat storage concept (GridReserve) to enable the reactor to supplement intermittent renewables. When electricity demand is low, the heat from a 300 MWe Stable Salt Reactor (SSR, see below) can be transferred to a nitrate salt held in storage tanks for up to eight hours, and later used to drive a turbine when demand rises. This heat storage technology is already used with concentrated solar power (CSP) but isn't suitable for conventional nuclear reactors, which produce heat at around 300°C; however, the SSR outlet temperature of about 600°C is high enough to be used with this system and give 900 MWe peaking capacity.

While MSR technology has been researched in many countries for decades, it is generally perceived that licensing MSRs is a major challenge and that in general there is so far very limited experience in design or operation of MSRs.

See also Molten Salt Reactors information paper for more detail of the designs described below.

MSRs with fuel in the primary salt coolant

Liquid Fluoride Thorium Reactor (LFTR)

The Liquid Fluoride Thorium Reactor (LFTR) is a heterogeneous MSR design which breeds its U-233 fuel from a fertile blanket of lithium-beryllium fluoride (FLiBe) salts with thorium fluoride. Some of the neutrons released during fission of the U-233 salt in the reactor core are absorbed by the thorium in the blanket salt. The resulting U-233 is separated from the blanket salt and in FLiBe becomes the liquid core fuel. LFTRs can rapidly change their power output, and hence be used for load-following.

Flibe LFTR

Flibe Energy in the USA is studying a 40 MW two-fluid graphite-moderated thermal reactor concept based on the 1960s-'70s US molten-salt reactor programme. It uses lithium fluoride/beryllium fluoride (FLiBe) salt as its primary coolant in both circuits. Fuel is uranium-233 bred from thorium in FLiBe blanket salt. Fuel salt circulates through graphite logs. Secondary loop coolant salt is sodium-beryllium fluoride (BeF_2-NaF). A 2 MWt pilot plant is envisaged, and eventually 600 MWt/250 MWe commercial plants.

Fuji MSR

The Fuji MSR is a graphite-moderated design to operate as a near-breeder with ThF_4-UF_4 fuel salt and FLiBe coolant at 700°C. It can consume plutonium and actinides, and be from 100 to 1000 MWe. It is being being developed internationally by a Japanese, Russian and US consortium: the International Thorium Molten Salt Forum (ITMSF) based in Japan. Several variants have been designed, including a 10 MWe mini Fuji. Thorium Tech Solutions (TTS) plans to commercialize the Fuji concept, and is working on it with the Halden test reactor in Norway.

Integral MSR

Canada-based Terrestrial Energy set up in 2013 has designed the Integral MSR (IMSR). This simplified MSR integrates the primary reactor components, including primary heat exchangers to secondary clean salt circuit, in a sealed and replaceable core vessel that has a projected life of seven years. The IMSR will operate at 600-700°C, which can support many industrial process heat applications. The moderator is a hexagonal arrangement of graphite elements. The fuel-salt is a eutectic of standard-assay (5%) low-enriched uranium fuel (UF_4) and a fluoride carrier salt at atmospheric pressure. Secondary loop coolant salt is ZrF_4-KF at atmospheric pressure. Tertiary steam is at 600°C for power generation, process heat, or to back up wind and solar. Emergency cooling and residual heat removal are passive. Each plant would have space for two reactors, allowing a seven-year changeover, with the used unit removed for offsite reprocessing when it has cooled and fission products have decayed. Terrestrial Energy hopes to commission its first commercial reactor in the 2020s.

The IMSR is scalable but from 2016 the company has been focused on a 440 MWt/195 MWe unit. The total levelized cost of electricity from the largest is projected to be competitive with natural gas. The smallest is designed for off-grid, remote power applications, and as prototype. Industrial heat at 600°C is also envisaged in 2016 plans. In September 2021 the company announced its 390 MWe IMSR400 upgraded power plant with twin reactors and generators.

Compared with other MSR designs, the company deliberately avoids using thorium-based fuels or any form of breeding, due to “their additional technical and regulatory complexities.” In September 2021 the company contracted Orano for full fuel services worldwide for the IMSR and in October it awarded contracts to BWXT Canada for steam supply systems.

In November 2017 Terrestrial Energy completed phase 1 of the Canadian Nuclear Safety Commission's (CNSC's) pre-licensing vendor design review of the IMSR-400, and in October 2018 it entered phase 2 of the review. In January 2019 the company notified the US Nuclear Regulatory Commission (NRC) of its intention to seek design approval for the IMSR-400. In December 2019 the CNSC and the US NRC selected Terrestrial Energy's IMSR for the first joint technical review of an advanced, non-light water nuclear reactor. Terrestrial Energy hopes to commission its first commercial reactor in the 2020s. The IMSR is a candidate for the US Advanced Reactor Demonstration Program but did not get a grant for early (seven-year) development.

In February 2019 the project progressed to stage 2 of site evaluation by Canadian Nuclear Laboratories – a separate process to licensing – in relation to possibly siting a commercial plant at Chalk River by 2026. Since November 2019 IMSR development has been supported by Canadian Nuclear Laboratories' Canadian Nuclear Research Initiative (CNRI). In October 2020 a C$20 million grant from Canada's Strategic Innovation Fund was announced, to accelerate development of the IMSR.

In January 2015 the company announced a collaborative agreement with US Oak Ridge National Laboratory (ORNL) to advance the design over about two years, and in May a similar agreement with the Dalton Nuclear Institute in the UK. In March 2017 the company entered into a contract with the University of New Brunswick for validation and verification work for the IMSR. In August 2021 the company signed an agreement with Westinghouse in the UK for fuel development and supply. The company has applied for a US loan guarantee of up to $1.2 billion to support financing of a project to license, construct and commission the first US IMSR, a 190 MWe commercial facility. In November 2021 the DOE made a $3 million grant to support licensing and commercialization of the IMSR.

Terrestrial Energy reviewed four potential US sites for the reactor, including one at Idaho National Laboratory (INL), and an agreement was signed with Energy Northwest in March 2018 for the first IMSR to be built here. The other three sites are located east of the Mississippi.

MicroNuclear molten salt battery

MicroNuclear LLC is developing what it calls a molten salt nuclear battery (MsNB). This is a concept for a small nuclear fission source providing heat by molten salt with no pumps or valves to power a commercial gas turbine of 5-10 MWe. No refuelling would be required for about ten years. The whole MsNB would be 3m diameter and 3m high. No other details. Idaho National Laboratory and Idaho University are involved.

Transatomic Power

Transatomic Power (TAP) is a US company partly funded by Founders Fund that initially aimed to develop a single-fluid MSR using very low-enriched uranium fuel (1.8%) or the entire actinide component of used LWR fuel. However, the company had to withdraw some exaggerated claims concerning actinide burn-up made in *MIT Technology Review* in 2016 and revised the design to using 5% enriched uranium. The revised TAP reactor design has a very compact core consisting of an efficient zirconium hydride moderator and lithium fluoride (LiF) based salt bearing uranium tetrafluoride (UF_4) fuel as well as the actinides that are generated during operation. The secondary coolant is FLiNaK (LiF-KF-NaF) salt to a steam generator. The neutron flux is greater than with a graphite moderator, and therefore contributes strongly to burning of the generated actinides. Fission products would be continuously removed while small amounts of fresh fuel added, allowing the reactor to remain critical for decades. Decay heat removal is by natural convection via a cooling stack.

A commercial reactor would be 1250 MWt/550 MWe running at 44% thermal efficiency with 650°C in the primary loop, using a steam cycle.

In September 2018 the company announced that it would cease operations and make its intellectual property freely available online.

ThorCon

Martingale in the USA is designing the ThorCon MSR (TMSR), which is a 250 MWe scaled-up Oak Ridge MSRE. It is a single-fluid thorium converter reactor in the thermal spectrum, graphite moderated. It uses a combination of U-233 from thorium and low-enriched U-235 (19.7% enriched) from mined uranium. Fuel salt is sodium-beryllium fluoride (BeF_2-NaF) with dissolved uranium and thorium tetrafluorides (Li-7 fluoride is avoided for cost reasons). Secondary loop coolant salt is also sodium-beryllium fluoride. It operates at 700°C. There is no online processing – this takes place in a centralized plant at the end of the core life – with off-gassing of some fission products meanwhile.

Several 550 MWt, 250 MWe TMSR modules would comprise a power station. Each module contains two replaceable reactors in sealed 'cans'. Each can contains a reactor ‘pot’, a primary heat exchanger and a primary loop pump. Each can is 11.6m high, 7.3m diameter and weighs 360 tonnes. The cans sit in silos below grade (30 m down). Below each is a 32-cylinder fuel salt drain tank, under a freeze valve.

At any one time, just one of the cans of each module is producing power. The other can is in cool-down mode. Every four years the can that has been cooling is removed and replaced with a new can. The fuel salt is transferred to the new can, and the can that has been operating goes into cool-down mode. In October 2015 Martingale signed an agreement with three Indonesian companies to commission a 500 MW ThorCon plant (TMSR-500) there. In 2020 Thorcon International was working with South Korea’s Daewoo Shipbuilding and Marine Engineering to build the TMSR500 as the first nuclear power plant (PLTN) in Indonesia.

In July 2020 Thorcon International signed a cooperation agreement with Indonesia’s Defence Ministry to evaluate developing a small TMSR (under 50 MW) for either power generation or marine propulsion. Thorcon will provide technical support for the ministry’s R&D.

Moltex SSR

Moltex Energy’s Stable Salt Reactor (SSR) is a conceptual UK MSR reactor design that relies on convection from static vertical fuel tubes in the core to convey heat to the reactor coolant. Because the nuclear material is contained in fuel assemblies, standard industrial pumps can be used for the low radioactivity coolant salt. Core temperature is 500-600°C, at atmospheric pressure. Decay heat is removed by natural air convection.

Fuel tubes three-quarters filled with the molten fuel salt are grouped into fuel assemblies which are similar to those used in standard reactors, and use similar structural materials. The fuel salt is about 60% NaCl, 20% $PuCl_2$, 20% UCl_3, with almost any level of actinide & lanthanide trichlorides mixed in depending on the spent oxide fuel used in reprocessing – about 16% fissile overall. The individual fuel tubes are vented so that noble fission product gases escape into the coolant salt, which is a ZrF_4-KF-NaF mixture, the radionuclide accumulation of which is managed. Iodine and caesium stay dissolved in the fuel salt. Other fission product gases condense on the upper fuel tube walls and fall back into the fuel mixture before they can escape into the coolant. The fuel assemblies can be moved laterally without removing them. Refuelling is thus continuous online, and after the fuel is sufficiently burned up the depleted assemblies are stored at one side of the pool for a month to cool, then lifted out so that the salt freezes. Reprocessing is straightforward, and any level of lanthanides can be handled.

SSR factory-produced modules are 150 MWe containing fuel, pumps, primary heat exchanger, control blades and instrumentation. Several, up to gigawatt-scale, can share a reactor tank, half-filled with the coolant salt which transfers heat away from the fuel assemblies to the peripheral steam generators, essentially by convection, at atmospheric pressure. There are three variants of the SSR: the Stable Salt Reactor – Wasteburner (SSR-W) fast reactor; about two years behind developmentally, the SSR-U thermal-spectrum reactor for a variety of applications; and the SSR-Th with thorium fuel. The GridReserve version has heat storage.

The SSR-W is the simplest and cheapest, due to compact core and no moderator. The primary fissile fuel in this original fast reactor version was to be plutonium-239 chloride with minor actinides and lanthanides, recovered from LWR fuel or from an SSR-U reactor. In 2020 the SSR-W fuel was 25% reactor-grade $PuCl_3$ with 30% UCl_3 and 45% KCl. Primary coolant salt is ZrF_4-KF at a maximum temperature of 590°C. Secondary coolant is nitrate salt buffer. Burn-up is 120-200 GWd/t. A 750 MWt/300 MWe demonstration plant is envisaged, the SSR-W300. An agreement has been signed with New Brunswick Power for initial deployment at Point Lepreau in Canada and in March 2021 the Canadian government announced a C$50.5 million investment towards this. In April 2021 plans were confirmed for this plus a plant for recycling used Canadian nuclear fuel for it. In November 2020 the two companies were joined by ARC Canada in setting up an SMR vendor cluster there. The Canadian Nuclear Safety Commission pre-licensing vendor design review of the SSR-W has completed the first phase. The first operating reactor is envisaged after 2030.

The company has announced the physically larger and more expensive SSR-U 'global workhorse version' of its design, with a thermal neutron spectrum running on LEU fluorides (up to 7% enriched) with graphite built into the fuel assemblies, which increases the size of the core. It runs at a higher temperature than the fast version – minimum 600°C – with ZrF_4-NaF coolant salt stabilized with ZrF_2. As well as electricity, hydrogen production is its purpose. It is designed to be compatible with thorium breeding to U-233. It is seen as having a much larger potential market, and initial deployment in the UK in the 2030s is anticipated, with potential for replacing CCGT and coal plants.

The SSR-Th is a thorium breeder version of the SSR-U, with thorium in the coolant salt and the U-233 produced is progressively dissolved in bismuth at the bottom of the salt pool. This contains U-238 to denature it and ensure there is never a proliferation risk. Once the desired level of U-233 is achieved (under 20%), the bismuth with uranium is taken out batch-wise, and the mixed-isotope uranium is chlorinated to become fuel. If the fuel is used in a fast reactor, plutonium and actinides can be added.

Moltex has also put forward its GridReserve molten nitrate salt heat storage concept to enable the reactor to supplement intermittent renewables.

Molten Chloride Fast Reactor

Southern Company Services in the USA is developing a molten chloride fast reactor (MCFR) with TerraPower, Oak Ridge National Laboratory (ORNL) – which hosts the work – the Electric Power Research Institute (EPRI) and Vanderbilt University. No details are available except that fuel is in the salt, and there is nothing in the core except the fuel salt. As a fast reactor it can burn U-238, actinides and thorium as well as used light water reactor fuel, requiring no enrichment apart from initial fuel load (these details from TerraPower, not Southern). It is reported to be large. The only other reactors using chloride fuel salts are the Elysium MCSFR and Moltex SSR.

In January 2016 the US DOE awarded a Gateway for Accelerated Innovation in Nuclear (GAIN) grant to the project, worth up to $40 million. In August 2016 Southern Nuclear Operating Company signed an agreement to work with X-energy to collaborate on development and commercialization of their respective small reactor designs. With TerraPower and ORNL, X-energy is designing the Xe-100 pebble-bed HTR of 48 MWe and the small Xe-Mobile microreactor.

In December 2020 the DOE selected Southern Company for a cost-share project of $113 million over seven years (DOE share $90 million) to develop the Molten Chloride Reactor Experiment (MCRE). This is a project to build a 300 kWt pool-type reactor to provide data and operational experience to inform the design, licensing, and operation of a demonstration MCFR based on TerraPower's technology. In November 2021 Southern and DOE signed an agreement to construct the MCRE at Idaho National Laboratory (INL). Collaborators in the MCRE project are TerraPower, INL, Core Power, Orano Federal Services, EPRI and 3M Company. The MCRE is expected to be operational in 2026.

The MCFR is being promoted by Core Power in the UK for marine use. It will not require refuelling during its operational life. Core Power aims to partner with technology developers to enable deployment of the marine MSR, including amending maritime regulations for wide acceptance of m-MSR powered ships worldwide.

In November 2020 it announced an agreement to work with TerraPower, Southern Company and Orano USA to develop MSR technology in the USA under the Advanced Reactor Demonstration Program.

Elysium MCSFR

Elysium Industries in the USA and Canada has the Molten Chloride Salt Fast Reactor (MCSFR) design with fuel in the chloride salt. It operates below grade at near atmospheric pressure. Primary fuel salt and secondary salt convey heat to steam generators at 650°C. It is designed to load-follow. A range of sizes from 125 to 3000 MWt (50 MWe to 1200 MWe) are under consideration. Used fuel from light water reactors or depleted uranium with some plutonium can fuel it though in 2020 fuel was shown as $PuCl_3$ with fission products, or 15% HALEU. Selected fission products are removed online. Passive safety includes a freeze plug. It has negative temperature and void coefficients.

MOSART

Russia’s Molten Salt Actinide Recycler and Transmuter (MOSART) is a larger fast reactor fuelled only by transuranic (TRU) fluorides from uranium and MOX LWR used fuel. The 2400 MWt design has a homogeneous core of Li-Na-Be or Li-Be fluorides without graphite moderator.

See also information page on Molten Salt Reactors.

Seaborg Compact Molten Salt Reactor

Seaborg Technologies in Denmark (founded 2015) has a thermal-epithermal single fluid reactor design for a 50 MWt pilot unit Compact Molten Salt Reactor (CMSR) with a view to 250 MWt commercial modular units fuelled by spent LWR fuel and thorium. Fuel salt is Li-7 fluoride initially with uranium as fluoride. Later, thorium, plutonium and minor actinides as fluorides are envisaged as fuel, hence the reactor being called a waste burner. This is pumped through the graphite column core and heat exchanger. Fission products are extracted online. Secondary coolant salt is FLiNaK, at 700°C. Spent LWR fuel would have the uranium extracted for recycle, leaving plutonium and minor actinides to become part of the MSR fuel, with thorium. The company claims very fast power ramp time. High temperature output will allow application to hydrogen production, synthetic fuels, *etc*.

In March 2017 the public funding agency Innovation Fund Denmark made a grant to Seaborg to "build up central elements in its long-term strategy and position itself for additional investments required to progress towards commercial maturity." This is the first Danish investment into nuclear fission research since the country introduced a ban on nuclear power in 1985. In December 2020 the American Bureau of Shipping (ABS) issued a feasibility statement regarding the reactor’s use on barges, with 200-800 MWe per barge. This is the first stage in the ABS's five-phase New Technology Qualification process. Seaborg aims to deploy the first full-scale prototype power barge by 2025.

MSRs with solid fuel (fluoride high-temperature reactors)

Mark 1 Pebble Bed FHR

This was a pre-conceptual US design completed in 2014 to evaluate the potential benefits of fluoride high-temperature reactor (FHR) technology. A consortium including University of California Berkeley, Oak Ridge National Laboratory and Westinghouse designed it as a 236 MWt/100 MWe pebble-bed FHR, with annular core, operating at 700°C. It is designed for modular construction, and from 100 MWe base-load it is able to deliver 240 MWe with gas co-firing for peak loads. Fuel pebbles are 30 mm diameter, much less than gas-cooled HTRs. The project looked at how FHRs might be coupled to a Brayton combined-cycle turbine to generate power, design of a passive decay heat removal system, and the annular pebble bed core. The PB-FHR has negative void reactivity and passive decay heat removal.

AHTR/FHR

Research on molten salt coolant has been revived at Oak Ridge National Laboratory (ORNL) in the USA with the Advanced High-Temperature Reactor (AHTR).[16] This is a larger reactor using a coated-particle graphite-matrix TRISO fuel like that in the GT-MHR (see above section on the GT-MHR) and with molten fluoride (FLiBe) salt as primary coolant. While similar to the gas-cooled HTR it operates at low pressure (less than 1 atmosphere) and higher temperature, and gives better heat transfer than helium. The FLiBe salt is used solely as primary coolant, and achieves temperatures of 750-1000°C or more while at low pressure. This could be used in thermochemical hydrogen manufacture.

A small version of the AHTR/FHR is the SmAHTR, with 125 MWt thermal size matched to early process heat markets, or producing 50+ MWe. Operating temperature is 700°C with FLiBe primary coolant and three integral heat exchangers. It is truck transportable, being 9m long and 3.5m diameter. Fuel is 19.75% enriched uranium in TRISO particles in graphite blocks or fuel plates. Refuelling interval is 2.5 to 4 years depending on fuel configuration. Secondary coolant is FLiNaK to Brayton cycle, and for passive decay heat removal, separate auxiliary loops go to air-cooled radiators. Later versions are intended to reach 850-1000°C, using materials yet to be developed.

Reactor sizes of 1500 MWe/3600 MWt are envisaged, with capital costs estimated at less than $1000/kW.

Kairos Power FHR and Hermes

Kairos Power in the USA has designed a 320 MWt/140 MWe fluoride (FLiBe) salt-cooled high temperature reactor (KP-FHR) which it plans to build at the East Tennessee Technology Park at Oak Ridge, Tennessee, in collaboration with Oak Ridge National Laboratory. The reactor uses 19.75% enriched TRISO fuel in pebble form with online refuelling and operates at up to 650°C. Secondary circuit salt is ‘solar’ nitrate, feeding a steam generator. It has passive shutdown and decay heat removal. The prototype is the Hermes reduced-scale test reactor of 35 MWt, selected by the DOE in December 2020 for a $629 million programme over seven years (DOE share $303 million). In May 2021 the Tennessee Valley Authority (TVA) agreed to provide engineering, operations, and licensing support for the Hermes project. TVA holds an early site permit for the Clinch River site. In October 2021 Kairos submitted its preliminary safety analysis report to the NRC as part of its construction licence application for the $100 million Hermes demonstration unit which it plans to bring online in 2026.

Thorium Molten Salt Reactor

China is planning a 10 MWe thorium-breeding molten-salt reactor (Th-MSR or TMSR), essentially an LFTR, with 2025 target for operation at the Shanghai Institute of Nuclear Applied Physics (SINAP, under the China Academy of Sciences). This is also known as the fluoride salt-cooled high-temperature reactor (FHR). It has low-enriched TRISO fuel as pebble bed, FLiBe primary coolant at 650°C and FLiNaK secondary coolant. A 100 MWt demonstration pebble-bed plant with open fuel cycle is planned by about 2025. SINAP sees this design having potential for higher temperatures than MSRs with fuel salt.

China claims to have the world's largest national effort on these and hopes to obtain full intellectual property rights on the technology. The US Department of Energy is collaborating with the China Academy of Sciences on the programme, which had a start-up budget of $350 million. The target date for TMSR deployment is 2032. See also US AHTR section above and information page on China's Nuclear Fuel Cycle.

Aqueous homogeneous reactors

Aqueous homogeneous reactors (AHRs) have the fuel mixed with the moderator as a liquid. Typically, low-enriched uranium nitrate is in aqueous solution. About 30 AHRs have been built as research reactors and have the advantage of being self-regulating and having the fission products continuously removed from the circulating fuel. A 1 MWt AHR operated in the Netherlands 1974-77 using Th-HEU MOX fuel. Further detail is in the Research Reactors paper.

A theoretical exercise published in 2006 showed that the smallest possible thermal fission reactor would be a spherical aqueous homogenous one powered by a solution of Am-242m$(NO_3)_3$ in water. Its mass would be 4.95 kg, with 0.7 kg of Am-242m nuclear fuel, and diameter 19 cm. Power output would be a few kilowatts. Possible applications are space program and portable high-intensity neutron source. The small size would make it easily shielded.

Heatpipe microreactors

Distinct from other small reactor designs, heatpipe reactors use a fluid in numerous sealed horizontal steel heatpipes to passively conduct heat from the hot fuel core (where the fluid vapourises) to the external condenser (where the fluid releases latent heat of vapourisation) with a heat exchanger. No pumps are needed to effect continuous isothermal vapour/liquid internal flow at less than atmospheric pressure. The principle is well established on a small scale, but here a liquid metal is used as the fluid and reactor sizes up to several megawatts are envisaged. There is a large negative temperature reactivity coefficient. There is very little decay heat after shutdown.

Experimental work on heatpipe reactors for space has been with very small units (about 100 kWe), using sodium as the fluid. They have been developed since 1994 at Los Alamos National Laboratory (LANL) as a robust and low technical risk system for space exploration with an emphasis on high reliability and safety, the Kilopower fast reactor being the best-known design.

Heatpipe microreactors may have thermal, epithermal or fast neutron spectrums, but above 100 kWe they are generally fast reactors.

It is generally perceived that licensing heatpipe reactors is a major challenge and that there is very limited or no experience in design or operation of them.

Westinghouse eVinci

The eVinci microreactor of 1 MWe to 5 MWe, but typically 1.6 MWe in present plans, would be fully factory built and fuelled. As well as power generation, process heat to 600°C would be available. Units would have a 40-year lifetime with three-year refuelling interval. They would be transportable, with setup under 30 days. The units would have 'walk-away' safety due to inherent feedback diminishing the nuclear reaction with excess heat, also effecting load-following. There are multiple fuel options for the eVinci, including uranium in oxide, metallic and silicide form. LANL and INL are researching the fuel. Westinghouse is aiming to complete the design, testing, analysis and licensing to build a demonstration unit by 2022, test by 2023, and have the eVinci ready for commercial deployment by 2025. In March 2020 the US Department of Defense awarded a contract for further development of the design (see Military developments section above), possibly using TRISO fuel, as the defense-eVinci (DeVinci), but $11.9 million DOD funding went only to March 2021. In December 2020 the DOE selected Westinghouse for a cost-share project of $9.3 million over seven years (DOE share $7.4 million) to develop the eVinci microreactor with a view to having a demonstration unit by 2024.

From October 2020 an agreement with Bruce Power in Ontario will assess the potential for off-grid deployment in Canada, where it has been submitted for CNSC pre-licensing vendor design review.

In March 2022 the Canadian government, through Innovation, Science and Economic Development Canada's (ISED's) Strategic Innovation Fund, announced investment of US $21.6 million in the eVinci reactor.

Oklo Aurora

Oklo Inc (formerly UPower) is a Californian company founded in 2013. It is developing a 1.5 MWe fast reactor using HALEU U-Zr metal fuel based on that in the EBR-II, but with lower burn-up. It is a heatpipe reactor with sealed heatpipes to convey heat from the reactor core to a supercritical carbon dioxide power conversion system to generate electricity. It is designed to operate for up to 20 years before refuelling. It is inherently safe, with a large temperature negative reactivity coefficient and does not require water cooling. It will be installed below grade. Idaho National Laboratory is working with the company on fuel and has agreed to host the prototype unit, for which the DOE has issued a site use permit. In June 2020 the US Nuclear Regulatory Commission accepted an application from Oklo for a combined construction and operating licence.

NuScale microreactor

In April 2019 NuScale announced that it was developing a 1-10 MWe "simple and inherently safe compact heat pipe cooled reactor" that "requires little site infrastructure, can be rapidly deployed, and is fully automated during power operation." Partners include Additech, INL, and Oregon State University. The project follows solicitation of ideas and designs from the US Department of Defense and the Department of Energy.

Others

LEADIR-PS100

This is a new design from Northern Nuclear Industries in Canada, combining a number of features in unique combination. The 100 MWt, 36 MWe reactor has a graphite moderator, TRISO fuel in pebbles, lead (Pb-208) as primary coolant, all as integral pool-type arrangement at near atmospheric pressure. It delivers steam at 370°C, and is also envisaged as an industrial heat plant. The coolant circulates by natural convection. The fuel pebbles are in four cells, each with graphite reflectors, and capacity can be increased by adding cells. Shutdown rods are similar to those in CANDU reactors. Passive decay heat removal is by air convection. The company presents it as a Gen IV design

Modular construction using small reactor units

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

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

Many small reactors are designed with a view to serial construction and collective operation as modules of a large plant. In this sense they are 'small modular reactors' – SMRs – but not all small reactors are of this kind (e.g. the Toshiba 4S), though the term SMR tends to be used loosely for all small designs.

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

Notes & references

Notes

a. In USA, UK, France, Russia, China, and India, mostly using high-enriched fuel. Reactors built as neutron sources are not designed to produce heat or steam, and are less relevant here. [Back]

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

c. Traditional reactor safety systems are 'active' in the sense that they involve electrical or mechanical operation on command. Some engineered systems operate passively, *e.g.* pressure relief valves. Both require parallel redundant systems. Inherent or full passive safety depends only on physical phenomena such as convection, gravity or resistance to high temperatures, not on functioning of engineered components. Because small reactors have a higher surface area to volume (and core heat) ratio compared with large units, a lot of the engineering for safety (including heat removal in large reactors) is not needed in the small ones. [Back]

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

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

Current-generation safety-related systems	SMR safety systems
High-pressure injection system. Low-pressure injection system.	No active safety injection system required. Core cooling is maintained using passive systems.
Emergency sump and associated net positive suction head (NPSH) requirements for safety-related pumps.	No safety-related pumps for accident mitigation; therefore, no need for sumps and protection of their suction supply.
Emergency diesel generators.	Passive design does not require emergency alternating-current (AC) power to maintain core cooling. Core heat removed by heat transfer through vessel.
Active containment heat systems.	None required because of passive heat rejection out of containment.
Containment spray system.	Spray systems are not required to reduce steam pressure or to remove radioiodine from containment.
Emergency core cooling system (ECCS) initiation, instrumentation and control (I&C) systems. Complex systems require significant amount of online testing that contributes to plant unreliability and challenges of safety systems with inadvertent initiations.	Simpler and/or passive safety systems require less testing and are not as prone to inadvertent initiation.

Current-generation safety-related systems	SMR safety systems
Emergency feedwater system, condensate storage tanks, and associated emergency cooling water supplies.	Ability to remove core heat without an emergency feedwater system is a significant safety enhancement.

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

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

Some of the early (1950s-1980) small power reactors were developed so as to provide an autonomous power source (ie not requiring continual fuel delivery) in remote areas. The USA produced eight such experimental reactors 0.3 to 3 MWe, deployed in Alaska, Greenland and Antarctica. The USSR produced about 20, of many kinds, and one (Gamma) still operates at the Kurchatov Institute. Another is the Belarus Pamir, mentioned in the HTR section above. [Back]

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

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

g. Central Argentina de Elementos Modulares (CAREM). See the Invap website (www.invap.com.ar). [Back]

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

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

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

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

The 300 MWe **THTR** (Thorium HochTemperatur Reaktor) in Germany was developed from the AVR and operated between 1983 and 1989 with 674,000 pebbles, over half containing Th/HEU fuel (the rest graphite moderator and some neutron absorbers). These were continuously recycled and on average the fuel passed six times through the core. Fuel fabrication was on an industrial scale. The reactor was shut down for sociopolitical reasons, not because of technical difficulties, and the basic concept with inherent safety features of HTRs was again proven. It drove a steam turbine.

The 200 MWt (72 MWe) **HTR-modul** was then designed by Siemens/Interatom as a modular unit to be constructed in pairs, with a core height three times its diameter, allowing passive cooling for removal of decay heat, eliminating the need for emergency core cooling systems. It was licensed in 1989, but was not constructed. This design was part of the technology bought by Eskom in 1996 and is a direct antecedent of the pebble bed modular reactor (PBMR).

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

In addition to these pebble bed designs, the 20 MWt Dragon reactor ran in UK 1964-75, the 115 MWt Peach Bottom reactor in USA ran 1966-74, and 8432 MWt Fort St Vrain ran 1976-89 – all with prismatic fuel, and the last two supplying power commercially. In the USA the Modular High-Temperature Gas-cooled reactor (MHTGR) design was developed by General Atomics in the 1980s, with inherent safety features, but the DOE project ended in 1993. [Back]

l. The 80 MWt ALLEGRO demonstration GFR is planned by Euratom to incorporate all the architecture and the main materials and components foreseen for the full-sized GFR but without the direct (Brayton) cycle power conversion system. It is being developed in a French-led project, and its preparatory phase is planned to 2026. [Back]

m. The Hyperion Power Module was originally designed by Los Alamos National Laboratory as a 70 MWt 'nuclear battery' that uses uranium hydride (UH3) fuel, which also functions as a moderator. UH3 stores vast quantities of hydrogen, but this stored hydrogen dissociates as the temperature rises above the operating temperature of 550°C. The release of hydrogen gas lowers the density of the UH3, which in turn decreases reactivity. This process is reversed as the core temperature drops, leading to the reabsorption of hydrogen. The consequent increase in moderator density results in an increase in core reactivity[11]. All this is without much temperature change since the main energy gain or loss is involved in phase change. [Back]

n. In October 2010, GEH announced it was exploring the possibility with Savannah River Nuclear Solutions of building a prototype PRISM reactor at the Department of Energy's Savannah River Site.

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

p. Most Air Cooled Condenser (ACC) technology has a limitation in that the tubes carrying the steam must be made of carbon steel which severely limits the service life of the ACC. Holtec has developed an ACC with stainless steel tubes bonded to aluminum fins and thus with much longer service life. [Back]

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Postscript/Appendix

Some of the developments described in this paper are fascinating and exciting. Nevertheless it is salutary to keep in mind the words of the main US pioneer in nuclear reactor development. Admiral Hyman Rickover in 1953 – about the time his first test reactor in the USA started up – commented on the differences between an "academic reactor" and a "practical reactor". See: http://en.wikiquote.org/wiki/Hyman_G._Rickover for the full quote:

An academic reactor or reactor plant almost always has the following basic characteristics: (1) It is simple. (2) It is small. (3) It is cheap. (4) It is light. (5) It can be built very quickly. (6) It is very flexible in purpose. (7) Very little development will be required. It will use mostly 'off-the-shelf' components. (8) The reactor is in the study phase. It is not being built now.

On the other hand a practical reactor can be distinguished by the following characteristics: (1) It is being built now. (2) It is behind schedule. (3) It is requiring an immense amount of development on apparently trivial items. (4) It is very expensive. (5) It takes a long time to build because of the engineering development problems. (6) It is large. (7) It is heavy. (8) It is complicated.

The tools of the academic-reactor designer are a piece of paper and a pencil with an eraser. If a mistake is made, it can always be erased and changed. If the practical-reactor designer errs, he wears the mistake around his neck; it cannot be erased. Everyone can see it.

Advanced Nuclear Power Reactors

(Updated April 2021)

- **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 page 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 page 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 information page) – 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 – EU HPR1000 – joined them in 2020 in meeting EUR.

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 Programme (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 page.

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	Operating at Shin Kori 3&4 in South Korea and at Barakah in UAE. Under construction: Shin Hanul 1&2 in South Korea. Korean design certification 2003. US design certification August 2019.
Gidropress	VVER-1200 (PWR)	1200	Operating at Novovoronezh II and Leningrad II in Russia, and at Ostrovets in Belarus. 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; two under construction in the USA; many units planned in China (as CAP1000).
Framatome (& EDF)	EPR (PWR)	1750	Two units operating in China, under construction in Finland, France and UK.
CNNC & CGN	Hualong One (PWR)	1170	Main Chinese export design, operating at Fuqing in China, and at Karachi in Pakistan.

Other advanced power reactors under construction

Developer	Reactor	Size – MWe gross	Design progress, notes
Gidropress	VVER-TOI (PWR)	1255	Under construction at Kursk II, planned for Nizhny Novgorod and many more in Russia.

INET & CNEC (China)	HTR-PM, HTR-200 module	2x105 (one module)	Demonstration plant being built at Shidaowan.
SNPTC	CAP1400/Guohe One	1500	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	Originally designed 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.
OKBM	VVER-600	600	Planned for Kola.

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 with EdF developed 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

SNPTC and SNERDI in China have jointly developed a passively safe 1500 MWe (4040 MWt) two-loop design from the AP1000, the CAP1400, or Guohe One, 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 started without public announcement in 2019. Exports are intended.

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 quotes 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 operating lifetime 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 is the ninth evolution of the original BWR design licensed in 1957, and 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 operating lifetime 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/V-392 units with enhanced safety (AES-92 & -91 power plants) have been built in India and China. Two more (V-466B variant) were planned for Belene in Bulgaria. The **AES-92** is certified as meeting EUR. The V-392 has four coolant loops, 163 fuel assemblies, and is rated 3000 MWt.

AES-2006, MIR-1200

The third-generation AES-2006 plant with **VVER-1200** (V-392M or V-491) reactors of 3212 MWt is an evolutionary development of the AES-92 and AES-91 plants with the VVER-1000, with longer operating lifetime (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 were being built at Novovoronezh II (V-392M) and Leningrad II (V-491), the first one starting operation in 2016. The first of two V-491 units at Ostrovets in Belarus commenced operation in 2020. Units based on the V-392M are being built Akkuyu in Turkey (V-509) and Rooppur in Bangladesh (V-523). The single V-522 in AES-2006E at Hanhikivi in Finland is based on the V-491.The Novovoronezh units provide 1114 MWe net each, and the Leningrad II units 1085 MWe net each, with a capacity factor of 90%. Two steam turbines are offered: Power Machines (Silmash) full-speed; and Alstom Arabelle half-speed, as proposed for MIR-1200 and Hanhikivi in Finland.

Overnight capital cost was said to be $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.

While OKB 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 page 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 3312 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 being built at Kursk II and planned for Smolensk II in Russia.

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

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 page.

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 page.

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.

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.

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 information page.

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 page.

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 Atomenergoproekt 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 page 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 page on Fast Neutron Reactors.

Generation IV designs

See information page on six Generation IV Reactors.

Small reactors

See also information page 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.

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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 UF6, 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 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.

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.