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Status of Small and Medium Sized Reactor Designs

A Supplement to the IAEA Advanced Reactors Information System (ARIS)
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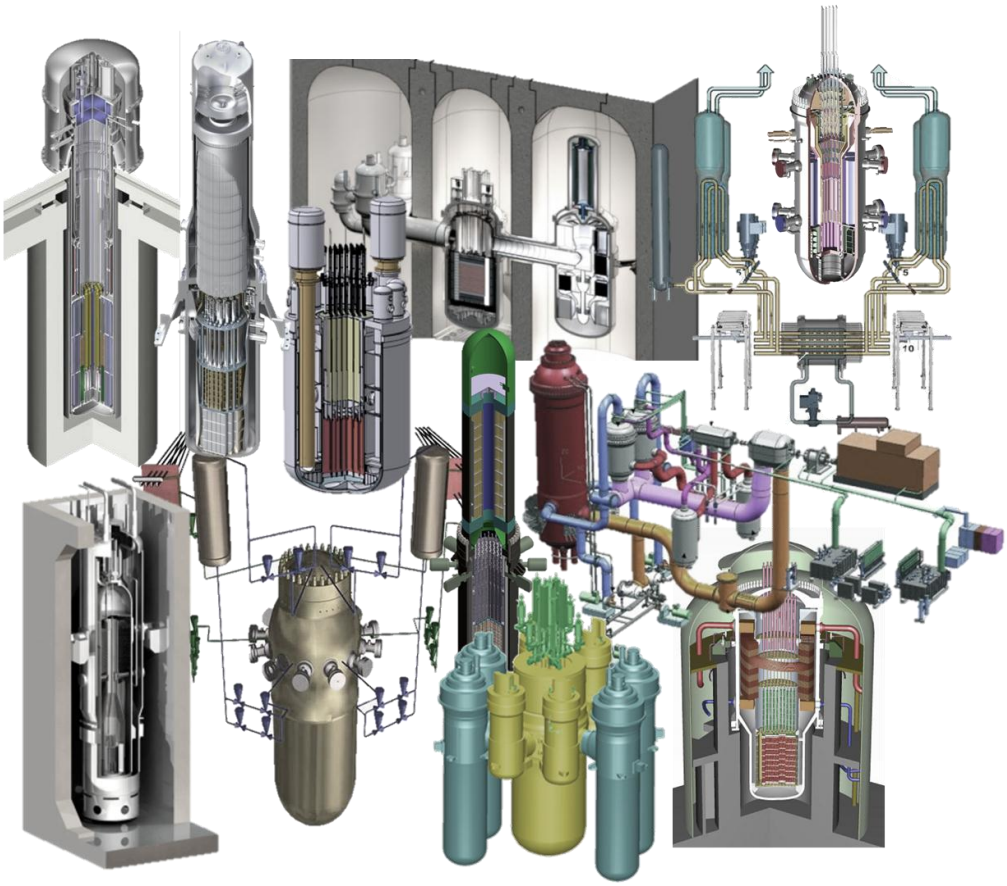

Nuclear Power Technology Development Section
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FOREWORD

There is renewed interest in Member States in the development and application of small and medium sized reactors (SMRs) having an equivalent electric power of less than 700 MW(e), or even less than 300 MW(e). At present, most new nuclear power plants under construction or in operation are large, evolutionary designs with power levels of up to 1700 MW(e), building on proven systems while incorporating technological advances. The considerable development work on small to medium sized designs generally aims to provide increased benefits in the areas of safety and security, non-proliferation, waste management, and resource utilization and economy, as well as to offer a variety of energy products and flexibility in design, siting and fuel cycle options. Specifically, SMRs address deployment needs for smaller grids and lower rates of increase in demand. They are designed with modular technology pursuing economies of series production, factory fabrication and short construction times. The projected timelines of readiness for deployment of SMR designs generally range from the present to 2025–2030.

The objective of this booklet is to provide Member States, including those considering initiating a nuclear power programme and those already having practical experience in nuclear power, with a brief introduction to the IAEA Advanced Reactors Information System (ARIS) by presenting a balanced and objective overview of the status of small and medium sized reactor designs.

This report is intended as a supplementary booklet to ARIS, which can be accessed at http://aris.iaea.org.

CONTENTS

INTRODUCTION

The ongoing interest in the development and deployment of reactors classified as small or medium sized is reflected in the number of small and medium sized reactors (SMRs) that operate or are under development and the numerous innovative concepts being investigated for electricity generation and for non-electrical applications. According to the classification adopted by the IAEA, small reactors are reactors with an equivalent electric power of less than 300 MW(e) and medium sized reactors are reactors with an equivalent electric power of between 300 MW(e) and 700 MW(e). Worldwide, 131 SMR units operate in 25 Member States, with a capacity of 63 GW(e). At present, 13 SMRs are under construction in six countries: Argentina, China, India, Pakistan, the Russian Federation and Slovakia. Research is being carried out on approximately 45 innovative SMR concepts for electricity generation and process heat production, desalination, hydrogen generation and other applications. SRMs are under development for all principal reactor lines, that is light water reactors (LWRs), heavy water reactors (HWRs), gas cooled reactors (GCRs) and liquid metal cooled reactors (LMCRs).

Small and medium sized LWRs are under development in Argentina, Japan, the Republic of Korea, the Russian Federation, the United States of America, France and Brazil. In Argentina, the Central Argentina de Elementos Modulares (CAREM) reactor, a small, integral type pressurized LWR design with all primary components located inside the reactor vessel and an electrical output of 150–300 MW(e), is under development. Construction of a 27 MW(e) CAREM prototype plant is planned to begin in 2012. In Japan, a 350 MW(e) integrated modular water reactor (IMR) suitable for a hybrid heat transport system with a natural circulation system is in the conceptual design stage. The System Integrated Modular Advanced Reactor (SMART) design from the Republic of Korea has a thermal capacity of 330 MW(th), is intended for seawater desalination and has almost reached the final design approval stage. In the Russian Federation, six light water SMR designs are under development. The ABV-6M, with an electrical output of 8.6 MW(e), is a nuclear steam generating plant with an integral pressurized light water reactor with natural circulation of the primary coolant; it is in the detailed design stage. The RITM-200 is designed to provide 8.6 MW(e). It is an integral reactor with forced circulation for universal nuclear icebreakers. The VK-300 is a 250 MW(e) simplified water cooled and water moderated boiling water reactor (BWR) with natural circulation of coolant and passive systems. The VBER-300 is a 325 MW(e) pressurized water reactor (PWR) conceptual design that can serve as a power source for floating nuclear power plants (NPPs). In addition, the Russian Federation is building two units of the KLT-40S series, to be mounted on a barge and used for cogeneration of process heat and electricity. Another Russian design is the pressurized water reactor UNITHERM, which is at the conceptual stage and is based on the N.A. Dollezhal Research and Development Institute of Power Engineering (NIKIET) design experience in marine nuclear installations.

In the USA, three integral pressurized water SMRs are under development: Babcock and Wilcox's mPower, NuScale and the Westinghouse SMR. The mPower design consists of four 125 MW(e) modules, and its design certification application is expected to be submitted in the fourth quarter of 2012. NuScale Power envisages an NPP made up of twelve 45 MW(e) modules and plans to apply for design certification with the US Nuclear Regulatory Commission (NRC) in 2012. The Westinghouse SMR is a conceptual design with an electrical output of 225 MW(e) incorporating passive safety systems and proven components of the AP1000. Another effort comes from the IRIS International Consortium, which is designing the International Reactor Innovative and Secure (IRIS), an integral PWR design with an electrical capacity of 335 MW(e). The Fixed Bed Nuclear Reactor (FBNR) is a Brazilian conceptual design that does not require on-site refuelling. The Flexblue design under development in France is a small seabed NPP with an output of between 50 and 250 MW(e).

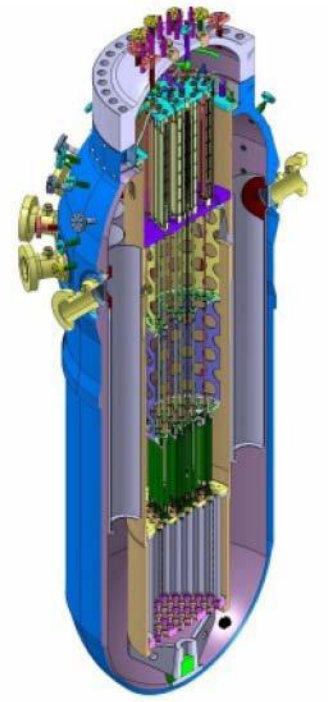
Heavy water SMRs are deployed in Argentina, Canada, China, India, the Republic of Korea, Pakistan, and Romania. Canada has developed and deployed the Canada deuterium–uranium reactor (CANDU) series, which offers various power ratings. The Enhanced CANDU 6 (EC6) is a basic design with a gross electrical capacity of 740 MW(e) that is based on the CANDU 6 design. In India, several HWRs ranging in size from 220 MW(e) to 540 MW(e) to 700 MW(e) are under construction or in operation. The 304 MW(e) Advanced Heavy Water Reactor with Low Enriched Uranium and Thorium Mixed Oxide Fuel (AHWR300-LEU) design incorporates vertical pressure tubes, low enriched uranium and thorium fuel, and passive safety features, and is in the basic design phase.

With regard to gas cooled reactors (GCRs), several designs in the SMR classification are under development in China, South Africa and the USA. China has developed, constructed and deployed the HTR-10, an experimental pebble bed helium cooled HTR. As a follow-up plant, in April 2011 China began construction of the HTR pebble bed module (HTR-PM) consisting of two 250 MW(th) modules. In South Africa, the Pebble Bed Modular Reactor (PBMR) conceptual design is a high temperature gas cooled reactor (HTGR) with an electrical output of 165 MW(e). In the USA, the 150 MW(e) gas turbine modular helium reactor is a conceptual design that has the potential to produce hydrogen by high temperature electrolysis or thermochemical water splitting. Finally, the energy multiplier module (EM^2) design is an effort to utilize used nuclear fuel without conventional reprocessing.

A number of liquid metal cooled SMRs have been designed and operated in China, France, India, Japan, the Russian Federation and the USA. The Chinese Experimental Fast Reactor (CEFR), a sodium cooled 20 MW(e) experimental fast reactor with PuO_2–UO_2 fuel, is currently in operation. It has reached criticality and is in the process of being commissioned. India is building the 500 MW(e) Prototype Fast Breeder Reactor (PFBR), which is expected to be commissioned in 2012. Japan has developed the Super Safe, Small And Simple (4S) reactor, designed to provide 10–50 MW(e) as a very small nuclear reactor design that can be located in a sealed, cylindrical vault underground, with the building above the ground. The Russian Federation's 300 MW(e) design BREST-OD-300 is a lead cooled fast reactor that uses a two circuit heat transport system to deliver heat to a supercritical steam turbine. The Russian Federation has also developed, and plans to construct, several SVBR-100 units, a small fast reactor with lead–bismuth eutectic alloy as the coolant and a power output of 100 MW(e). Finally, in the USA, the Power Reactor Innovative Small Module (PRISM), a 155 MW(e) liquid metal cooled fast breeder reactor, has been developed and an application to the US NRC for design certification is being prepared. The Hyperion Power Module (HPM) design with an electrical power output of 25 MW(e) is in the conceptual design stage.

LIGHT WATER REACTORS

CAREM (CNEA, Argentina)



Full name: Central Argentina de Elementos Modulares
Reactor type: Integral pressurized water reactor
Coolant/moderator: Light water
System pressure: 12.25 MPa
System temperature: 326°C
Thermal capacity: 100 MW(th)
Design life: 60 years
Fuel material: UO_2
Fuel enrichment: 3.1%
Fuel cycle: 14 months
Design status: Detail engineering, licensing process for deployment
Distinguishing features: The power range of CAREM modules (150 MW(e) for natural circulation) allows sequencing of additional capacity to more closely match demand

CAREM (Central Argentina de Elementos Modulares) is a project of Argentina's National Atomic Energy Commission (CNEA) whose purpose is to develop, design and construct an innovative, simple and small NPP.

This integral type PWR has an indirect cycle with distinctive features that simplify the design and support the objective of achieving a higher level of safety. Some of the high level design characteristics of the plant are: an integrated primary cooling system, a self-pressurized primary system, in-vessel hydraulic control rod drive mechanisms and safety systems relying on passive features. For power modules below 150 MW(e), coolant flow in the primary reactor system is achieved by natural circulation.

Due to the integrated design approach, the pressurizer and steam generators (SGs) are located inside the reactor pressure vessel (RPV). The location of the SG above the core produces natural circulation in the primary circuit. The secondary system circulates upwards inside the SG tubes, while the primary system circulates in a countercurrent flow.

Due to self-pressurization, the core outlet, riser and dome temperatures are very close to the saturation temperature. Under all operating conditions, this has proved to be sufficient to guarantee remarkable stability of the reactor coolant system (RCS). The control system is capable of keeping the reactor pressure at the operating point during different transients, even in the case of power ramps. The negative reactivity feedback coefficients and the RCS large water inventory combined with the self-pressurization features make this behaviour possible with minimal control rod motion.

The defence in depth concept has been internalized in the design, in order to improve safety significantly compared with the current generation of NPPs. Many intrinsic characteristics contribute to the preclusion of typical LWR initiating events like large and medium loss of coolant accidents (LOCAs), loss of flow accidents, boron dilution and control rod ejection. CAREM safety systems are based on passive features; neither AC power nor operator actions are required to mitigate the postulated design events during the grace period (32 hours for each of the two redundancies). The safety systems are duplicated to fulfill the redundancy criteria, and the shutdown system is diversified to fulfill regulatory requirements.

The CAREM concept was presented in 1984 in Lima, Peru, during the IAEA conference on small and medium sized reactors [1]. CAREM was one of the first of the new generation reactor designs. The next step of the project is to start construction of the 100 MW(th) demonstration plant (CAREM-25) by 2012. CAREM has been recognized as an international near term deployment (INTD) reactor by the Generation IV International Forum (GIF).

CNP-300 (CNNC, China)

Full name: Chinese nuclear power unit
Reactor type: Pressurized water reactor
Coolant/moderator: Light water
System pressure: 15.2 MPa
System temperature: 302°C
Thermal capacity: 999 MW(th)
Electrical capacity: 325 MW(e)
Design life: 40 years
Fuel material: UO_2
Fuel enrichment: 2.4–3.0%
Fuel cycle: 12 months
Design status: In operation
Distinguishing features: Operating small two loop PWR

The CHASNUPP-1 is located on the Chashma site in the northwestern region of the Thal Doab in the Punjab province of Pakistan. C-1 is a single unit of 300 MW(e) class and includes a two loop PWR nuclear steam supply system (NSSS) furnished by the China National Nuclear Corporation (CNNC).

The systems and the major equipment of the nuclear island, including the NSSS, are designed by the Shanghai Nuclear Engineering Research and Design Institute (SNERDI), and the system of the conventional island is designed by the East China Electric Power Design Institute (ECEPDI). The NSSS is designed for a power output of 1002 MW(th). The corresponding gross electric output of the turbine generator is 325 MW(e) and the net output of the plant is around 300 MW(e). Accidents such as a LOCA, steam line break or other postulated accident having off-site dose consequences are also analysed at a power level of 1066 MW(th).

The reactor containment is a pre-stressed reinforced concrete structure in the form of a vertical cylinder with a torispherical dome and a flat base.

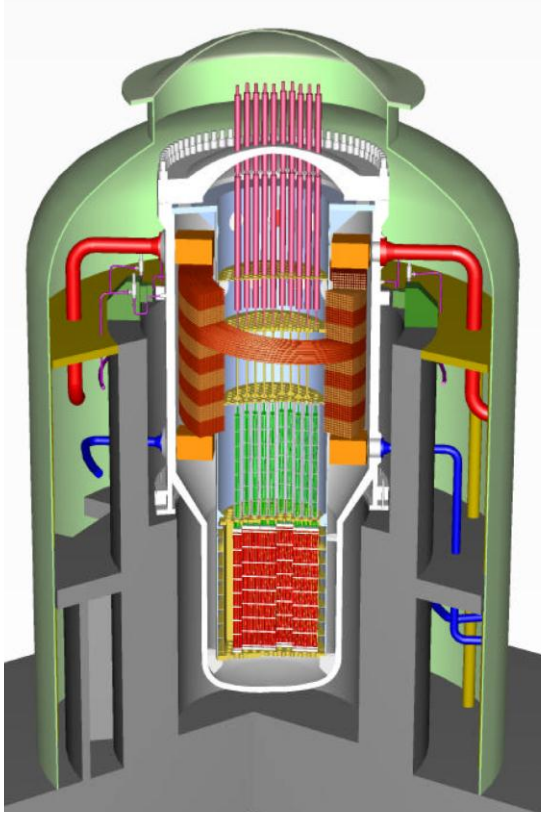
The reactor core consists of 121 fuel assemblies (FAs). Each FA is composed of uranium dioxide pellets enclosed in pressurized zircaloy tubes. Rod cluster control assemblies are used for reactor control and consist of clusters of cylindrical stainless steel clad silver–indium–cadmium absorber rods. The absorber rods move within guide tubes in certain FAs. The core is of the multi-enrichment region type. In the initial core loading, three kinds of fuel enrichment have been used.

The NSSS for the plant consists of a PWR, an RCS and associated auxiliary systems. The RCS is arranged as two closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and an SG. The reactor coolant pumps are vertical, single stage, axial flow pumps. High pressure water circulates through the reactor core to remove the heat generated by the nuclear chain reaction. The heated reactor coolant exits from the reactor vessel and passes via the coolant loop piping to the SG. Here it gives up its heat to the feedwater to generate steam for the turbine generator. The cycle is completed when the water from the SG is pumped back into the reactor vessel.

Several engineered safety features have been incorporated into the plant design to reduce the consequences of a LOCA. These safety features include a safety injection system. The safety injection system also serves to insert negative reactivity into the reactor core during an uncontrolled plant cooldown. Another safety feature is the containment spray system which also serves to remove airborne elemental iodine from the containment atmosphere following a LOCA. The reactor is refuelled by equipment that handles spent fuel under water from the time it is removed from the reactor vessel until it is placed in a shipping cask for shipment from the site. The fuel handling system is divided into two areas: the reactor cavity, which is flooded for refuelling; and the spent fuel storage pool, which is located in the fuel building and is always accessible to plant personnel. The two areas are connected by a fuel transfer system that carries the fuel through a fuel transfer tube between the fuel building and the containment.

C-1 recently completed ten years of operation and has undergone a preliminary safety review [2].

IMR (Mitsubishi Heavy Industries, Japan)



Full name: Integrated modular water reactor
Reactor type: Integral pressurized water reactor
Coolant/moderator: Light water
System pressure:15.5 MPa
System temperature: 345°C
Thermal capacity: 1000 MW(th)
Electrical capacity: 350 MW(e)
Design life: 60 years
Fuel material: UO_2
Fuel enrichment: 4.8%
Fuel cycle: 26 months
Design status: Conceptual design
Distinguishing features: Hybrid heat transport system, natural circulation system under bubbly flow conditions for primary heat transport

The IMR is a medium sized power reactor with a reference output of 350 MW(e) and an integral primary system with the potential for deployment after 2020.

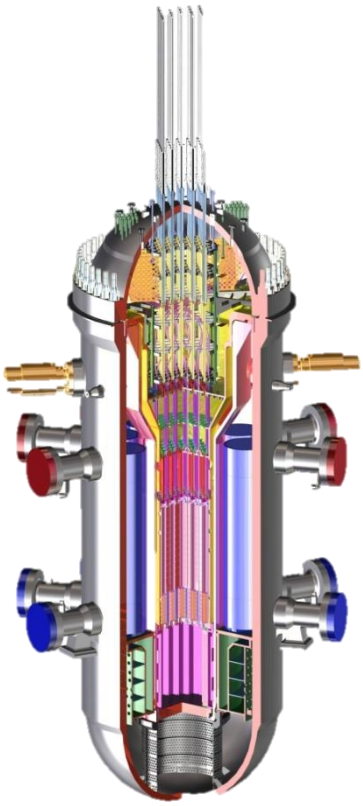
The IMR is an LWR with moderation ratios similar to those of conventional LWRs, and thus its properties of fresh and spent fuel are similar to those of LWRs. This similarity allows for the basic adoption of conventional safeguards procedures and LWR management practices for new and spent fuel. The IMR has no reactor coolant pumps, pressurizer or coolant pipes with large diameters, nor does it have an emergency core cooling system (ECCS) or containment cooling spray system. Simple support systems, such as the component cooling water system, the essential service water system and the emergency AC power system, are designed as non-safety grade systems, possible by use of a stand-alone diesel generator. Due to the integrated primary system, the containment vessel is small, and simplified chemical and volume control and waste disposal systems are the result of the boric acid free operation.

The IMR is primarily designed to generate electricity as a land based power station module. The capacity of the power station can be adjusted to meet demand by constructing additional modules. Because of its modular characteristics, it is suitable for large scale power stations consisting of several modules as well as for small distributed power stations, especially when the capacity of the grid is small. The IMR also has the capability for district heating, seawater desalination, process steam production and other non-electric applications.

The IMR conceptual design study was initiated in 1999 by Mitsubishi Heavy Industries (MHI) reflecting changes in the business environment. A group led by MHI and including Kyoto University, the Central Research Institute of the Electric Power Industry (CRIEPI), and the Japan Atomic Power Company (JAPC) developed related key technologies through two projects, funded by the Japan Ministry of Economy, Trade and Industry (METI), from 2001 to 2004 and from 2005 to 2007.

Validation testing, research and development (R&D) for components and design methods, and basic design development are required before licensing. The time required for development and deployment of the IMR depends on the financial situation and the extent of construction requirements; the target year to start licensing is 2020 at the earliest.

SMART (KAERI, Republic of Korea)



Full name: System integrated modular advanced reactor
Reactor type: Integral pressurized water reactor
Coolant/moderator: Light water
System pressure:15 MPa
System temperature: 323°C
Thermal capacity: 330 MW(th)
Electrical capacity: 100 MW(e)
Design life: 60 years
Fuel material: UO_2
Fuel enrichment: 4.8%
Fuel cycle: 36 months
Design status: Seeking standard design approval by 2011
Distinguishing features: Coupling of the desalination system or process heat application

SMART is a small integral PWR with a rated thermal power of 330 MW(th). It aims at achieving enhanced safety and improved economics; to enhance safety and reliability, the design organization has incorporated inherent safety features and reliable passive safety systems. The aim is to achieve improvement in the economics through system simplification, component modularization, reduction of construction time and high plant availability.

By introducing a passive residual heat removal system and an advanced LOCA mitigation system, significant safety enhancement is expected by the design organization. The low power density design with about a 5 wt% UO_2 fuelled core will provide a thermal margin of more than 15% to accommodate any design basis transients with regard to the critical heat flux. This feature ensures the core thermal reliability under normal operation and any design basis events.

SMART fuel management is designed to achieve a maximum cycle length between refuelling outages. A simple two batch refuelling scheme without reprocessing provides a cycle of 990 effective full power days for a 36 month operation. This reload scheme minimizes complicated fuel shuffle schemes and enhances fuel utilization. The SMART fuel management scheme is highly flexible to meet customer requirements.

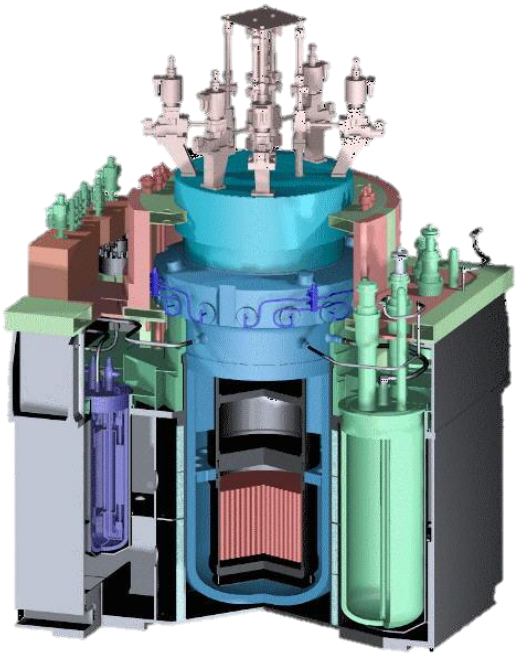
The design incorporates highly reliable engineered safety systems that are designed to function automatically. These consist of a reactor shutdown system, a safety injection system, a passive residual heat removal system, a shutdown cooling system and a containment spray system. Additional engineered safety systems include a reactor overpressure protection system and a severe accident mitigation system.

The secondary system receives superheated steam from the NSSS and uses most of the steam for electricity generation and pre-heaters, and the remainder for non-electric applications. The seawater desalination system may be used in conjunction with the secondary system.

Since 1997, the Government of the Republic of Korea has been supporting the development of the SMART technology, which was divided into three phases: technology development, technology verification and commercialization.

Commercialization of the SMART desalination plant will begin in 2013, just after the standard design approval phase.

ABV-6M (OKBM Afrikantov, Russian Federation)



Reactor type: Pressurized light water reactor
Coolant/moderator: Light water
System pressure: 15.7 MPa
System temperature: 330°C
Thermal capacity: 38 MW(th)
Electrical capacity: 8.6 MW(e)
Design life: 60 years
Fuel material: UO_2
Fuel enrichment: 19.7%
Reactor core life: 10 years
Design status: Detailed design
Distinguishing features: Natural circulation in the primary circuit for land based and floating nuclear power plants

The ABV-6M reactor installation is a nuclear steam generating plant with an integral pressurized LWR and natural circulation of the primary coolant. The ABV-6M design was developed using the operating experience of water cooled, water moderated power reactors (WWERs) and recent achievements in the field of NPP safety. The main objective of the project is to create small, multipurpose power sources based on proven marine reactor technologies, providing easy transport to the site, rapid assembly and safe operation.

The reactor is designed to produce 45 MW(th) and 8.6 MW(e) in condensation and 14 MW(th) and 6 MW(e) in cogeneration mode [3]. The reactor operates under the normal PWR conditions of 15.7 MPa in the RPV. The SGs located inside the vessel generate 290°C steam at 3.14 MPa flowing at 55 t per hour.

The reactor cover is set under biological shielding and the control rod drive mechanism (CRDM) is located above the shield outside the vessel.

The main reactor pump equipment is arranged on the shield tank as a single steam generating aggregate (SGA) which is transportable by rail. The SGA has a mass of 200 t and is 5 m long by 3.6 m wide with a height of 4.5 m.

The core lifetime without reloading or shuffling of fuel is 10 years with 16 000 hours of continuous operation. Specifically, the ABV reactor installation is intended as a universal power source for floating NPPs. The reactor is designed with the capability of driving a floating unit with a maximum length of 140 m, a beam of 21 m, a draft of 2.8 m and a displacement of 8700 t [3]. Depending on the needs of the region, the floating NPP can generate electric power or provide heat and power cogeneration or heat generation for seawater desalination, or can be used for other applications.

The stationary NPP — land based or underground — is fabricated as large ready-made units; these units are transported to the site by special truck or by water. The total reactor module has a mass of 600 t and is 13 m in length and 8.5 m in diameter. This allows for a small footprint for the plant building, which would be approximately 67 m long and 47 m wide. The floating NPP is factory fabricated.

RITM-200 (OKBM Afrikantov, Russian Federation)

Full name: RITM-200

Reactor type: Integral pressurized water reactor

Coolant/moderator: Light water

System pressure: 15.7 MPa

Thermal capacity: 175 MW(th)

Electrical capacity: 50 MW(e)

Design life: 40 years

Fuel material: UO_2

Fuel enrichment: <20%

Fuel cycle: 84 months

Design status: Detailed design

Distinguishing features: Integral reactor with forced circulation in the primary circuit for universal nuclear icebreakers

The RITM-200 is being designed by OKBM Afrikantov as an integral reactor with forced circulation for use in universal nuclear icebreakers.

It employs a low enriched cassette type reactor core similar in design to the KLT-40S. The fuel enrichment is up to 20 wt% and the reactor houses 199 FAs. The design also allows for a lower fluence on the reactor vessel.

The reactor is designed as an integral vessel with the main circulation pumps located in separate external hydraulic chambers and with side horizontal sockets for SG cassette nozzles. The four section SG has 12 rectangular cassettes, while the four main circulation pumps are located in the cold leg of the primary circulation path separated into four independent loops. The reactor is also designed to use forced circulation of the primary coolant and an external gas pressurizer system. The SGs produce steam of 295°C at 3.82 MPa flowing at 248 t per hour.

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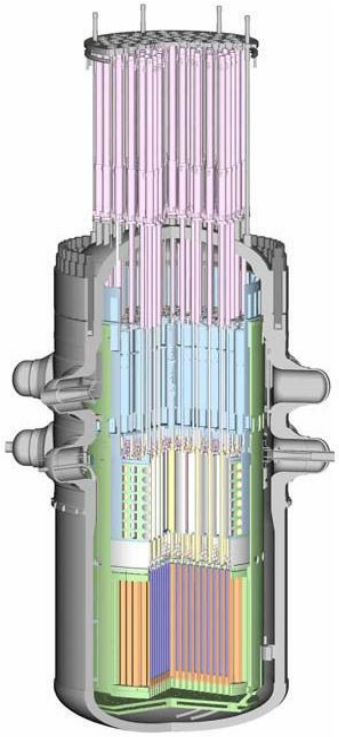
The core is designed to operate for 7 years at a 65% capacity factor before the need for refuelling. The assigned service life of the plant for replaceable equipment is 20 years with a continuous operation period of 26 000 h; for permanent equipment it is double that.

The RITM-200 is designed to provide 30 MW of shaft power on a typical nuclear icebreaker and can be used on vessels of 150–300 t displacement. The reactor can also be considered for floating heat and power plants, power and desalination complexes, and offshore drilling rigs. The designers also claim that the overall size of the steam generating unit allows transportation of the reactor by rail. The reactor plant in containment has a mass of 1100 t and is 6 m × 6 m × 15.5 m.

VK-300 (RDIPE, Russian Federation)



Reactor type: Boiling water reactor
Coolant/moderator: Light water
System pressure: 6.9 MPa
System temperature: 285°C
Thermal capacity: 750 MW(th)
Electrical capacity: 250 MW(e)
Design life: 60 years
Fuel material: UO_2
Fuel enrichment: 4%
Fuel cycle: 18 months
Design status: Conceptual design
Distinguishing features: Single circuit scheme and natural circulation of primary circuit

The VK-300 is a 250 MW(e) simplified water cooled and water moderated BWR with natural circulation of coolant and passive systems [1]. The major technological principles of the concept are: the simplicity of design resulting from a single circuit scheme and natural circulation of coolant in the core; the use of water as a coolant and moderator; the production of steam of the required parameters directly in the reactor due to an integral arrangement; and strong reliance on passive features and systems to achieve a high level of safety.

International achievements in the field of designing and operating boiling type reactors have been also taken into account, primarily as far as the design of passive safety systems. The VK-300 design features are based on the use of equipment and components developed and manufactured for other reactor types.

The reactor core is cooled by natural circulation during normal operation and in any emergency. The design reduces the mass flow rate of coolant by initially extracting moisture from the flow and returning it to the core inlet, ensuring a lower hydraulic resistance of the circuit and raising the natural circulation rate.

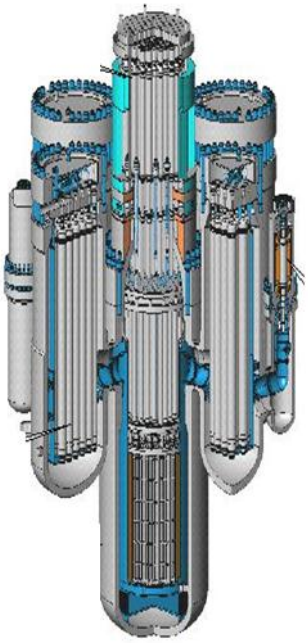
The VK-300 reactor has a small reactivity margin for nuclear fuel burnup thanks to partial overloading and use of burnable absorbers.

Another innovative feature of the VK-300 project is the application of a metal lined primary containment (PC) of reinforced concrete. The PC helps to solve the safety assurance problem economically and reliably using structurally simple passive safety systems.

The design of the 750 MW(th) and 250 MW(e) VK-300 was developed to supply electricity and heat of up to 465 MW(th) within a nuclear cogeneration plant to be built at the Krasnoyarsk Mining and Chemical Combine. This design was developed by Russian research and design organizations including the Research and Development Institute of Power Engineering (NIKIET), the Russian National Research Centre 'Kurchatov Institute' and the Institute of Physics and Power Engineering (Obninsk), and involving the Research Institute of Atomic Reactors (RIAR), A.A. Bochvar All-Russia Research Institute of Inorganic Materials (VNIINM), All-Russia Design and Research Institute for Integrated Power Technology (VNIPIET) and others.

Research and development activities are currently under way for further validation of the design approach adopted in the VK-300 design.

VBER-300 (OKBM Afrikantov, Russian Federation)



Reactor type: Pressurized water reactor
Coolant/moderator: Light water
System pressure: 12.7 MPa
System temperature: 316°C
Thermal capacity: 917 MW(th)
Electrical capacity: 325 MW(e)
Design life: Conceptual design
Fuel material: UO_2
Fuel enrichment: 4.95%
Fuel cycle: 72 months
Design status: Conceptual design
Distinguishing features: Power source for floating NPPs

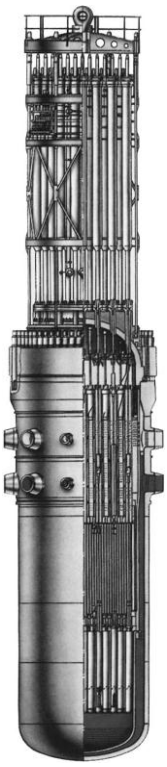
The VBER-300 reactor is a medium sized power source for land based NPPs and nuclear cogeneration plants, as well as for floating NPPs.
The VBER-300 design is a result of the evolution of modular marine propulsion reactors. The thermal power increase is due to an increase in mass and overall dimensions, while the reactor's appearance and main design solutions are kept as close as possible to those of marine propulsion reactors. The design is being developed using the operating experience of WWER type reactors and achievements in the field of NPP safety. Specific features of the VBER-300 are:

- Use of proven nuclear ship building technologies, engineering solutions and operating experience of WWER reactors;
- High level of safety;
- Economic competitiveness of VBER-300 based NPPs;
- Possibility to enlarge or reduce the source power using only unified VBER-300 equipment (reactors consisting of two, three or five loops).

The VBER-300 design concept allows a flexible fuel cycle for the reactor core with standard WWER FAs. The interval between partial refuelling is 1–2 years. The number of FAs in the refuelling batch is either 15 or 30; maximal fuel burnup does not exceed 60.0 MW·d/kg U for the cycle with 30 fresh FAs in the reloading batch and maximum initial uranium enrichment.

Light water acts as the primary coolant and moderator. The hot primary coolant is cooled in the once through SG that generates slightly superheated steam and directs it to the turbine. Part of the steam is taken from the turbine to heat up the district heating circuit fluid.
The safety assurance engineering solutions incorporated into the design are: prioritization of accident prevention measures, design simplification, inherent safety, and the defence in depth principle; passive safety systems and enhancement of safety against external impacts (including acts of terrorism); and limitation of severe accident consequences.
The VBER-300 preliminary design was completed in 2002, and a technical and commercial proposal (a shorter version of technical and economic investigation) for construction of a land based or floating NPP with the VBER-300 was prepared.
The preliminary design passed the branch review by Rosatom and was approved by the Scientific and Technical Council. Currently, there are two directions of further project development: first, within the framework of scientific and technical cooperation between the Russian Federation and Kazakhstan; and second, replacement of outdated NPP capabilities or construction of new medium sized NPPs in the Russian Federation.

WWER-300 (OKBM Gidropress, Russian Federation)



Reactor type: Pressurized water reactor
Coolant/moderator: Light water
System pressure: 16.2 MPa
System temperature: 325°C
Thermal capacity: 850 MW(th)
Electrical capacity: 300 MW(e)
Design life: Conceptual design
Fuel material: UO_2
Fuel enrichment: 4.95%
Fuel cycle: 72 months
Design status: Detailed design
Distinguishing features: Based on the operating experience of various WWER type reactor designs

The design of the two loop reactor with WWER-300 (V-478) is based on engineering solutions for the equipment of WWER design. The design of the V-407 is taken as a reference.

The design of the WWER-300 (V-478) is based on a design developed for small power grids, using the structure, materials and parameters of primary side equipment based on the WWER-640 (V-407) design and using FAs similar to those used in the WWER-100.

The reactor core comprises 85 FAs and 34 control and protection system control rods. Each FA contains 312 fuel rods with a maximum ^{235}U enrichment of 3.3%. The number of fresh FAs loaded into the core annually for the base fuel cycle is 24.

The reactor primary loop consists of a model PGV-640 horizontal SG, GTSNA-1455 reactor coolant pumps (RCPs), a pressurizer and all of the main coolant pipelines. The RCP is a vertical pump with a drive operated electrical motor with a flywheel and auxiliary systems.

The design was developed according to the requirements of the Russian Federation's current regulations, standards for atomic energy, IAEA safety standards and other recommendations, as well as the requirements of European operators of NPPs. The safety features used in the V-478 design include: SG emergency cooldown systems, emergency gas removal systems, a boron injection system, a main steam line isolation system, high and low pressure core cooling pumps and passive heat removal.

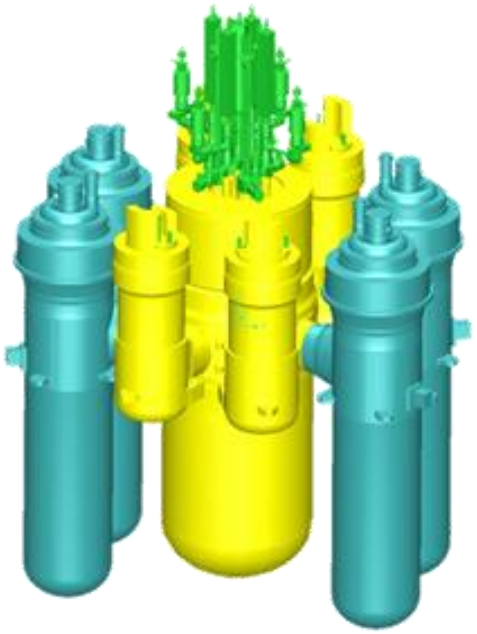
The feedwater system comprises the main feedwater pumps, standby feedwater pump, deaerator, isolation and control valves, and pipelines. The feedwater supply is provided by three feedwater pumps from the deaerator plant.

The composition and design of the main components, equipment and systems are based on existing designs, improved according to the up to date requirements that enable improved performance and ensure the required safety level. Basic technical solutions for the NPP are proved by the more than 1400 reactor-years of operating experience of WWER plants.

The NPP construction time from the initial stage to commissioning for commercial operation is expected by the designer to be 4.5 years.

R&D must be provided for the verification of fuel load follow conditions, instrumentation and control requirements, hydrogen safety for design basis accidents, the TVS-2M application in the WWER-300 and severe accident safety analyses.

KLT-40S (OKBM Afrikantov, Russian Federation)



Reactor type: Pressurized water reactor
Coolant/moderator: Light water
System pressure: 12.7 MPa
System temperature: 316°C
Thermal capacity: 150 MW(th)
Electrical capacity: 35 MW(e)
Design life: 40 years
Fuel material: UO_2
Fuel enrichment: <20%
Fuel cycle: 28 months
Design status: Under construction
Distinguishing features: Floating nuclear power plant

The KLT-40S is a pressurized water reactor developed for a floating NPP. It is based on the commercial KLT-40 marine propulsion plant and is an advanced variant of reactor plants that power nuclear icebreakers.

The reactor is a modular design with the core, SGs and main circulation pumps connected with short nozzles. The reactor has a four loop system with forced and natural circulation, a pressurized primary circuit with canned motor pumps and leaktight bellow type valves, a once through coiled SG and passive safety systems.

Fuel utilization efficiency is provided by the use of all improvements of nuclear fuel and fuel cycles of nuclear icebreaker reactors, spent fuel reprocessing and the increase of fuel burnup through the use of dispersion fuel elements.

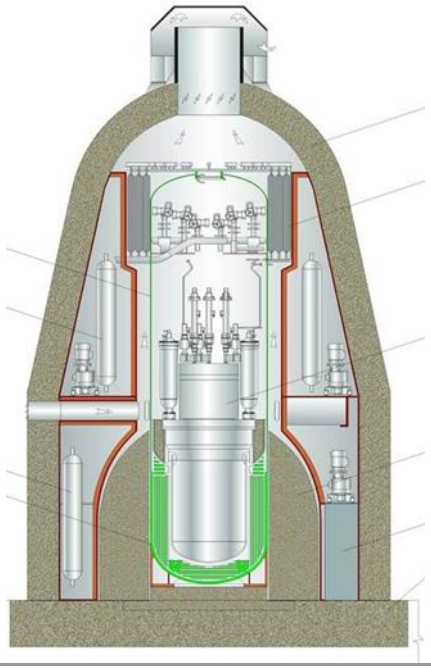
The floating NPP with KLT-40S reactor can be manufactured in shipyards and then delivered to the customer fully assembled, tested and ready for operation. There is no need to create transportation links, power transmission lines, or the preparatory infrastructure required for land based NPPs, and there is a high degree of freedom in selecting the location for a floating NPP as it can be moored in any coastal region. It also provides a short construction period of four years. The availability of the entire nuclear vessel servicing and maintenance infrastructure in the Russian Federation will permit costs to be minimized for floating NPP maintenance and refuelling.

One of the advantages foreseen by the floating power unit (FPU) based ATES-MM under construction is long term autonomous operation in remote regions with a decentralized power supply. The design requires that after every 3–4 years of operation the reactor be refuelled; the spent nuclear fuel is then stored on board the FPU and no special maintenance or refuelling ships are necessary.

The KLT-40S is designed with proven safety solutions such as a compact structure of the SG unit with short nozzles connecting the main equipment, without large diameter primary circuit pipelines, and with proven reactor emergency shutdown actuators based on different operation principles, emergency heat removal systems connected to the primary and secondary circuits, elimination of weak design points based on the experience of prototype operation and use of available experimental data, certified computer codes and calculation procedures.

Construction of the FPU and equipment fabrication has been under way since 2007. At present, fabrication of the main equipment for the ATES–MM reactor and turbine generator sets is nearing completion.

UNITHERM (RDIPE, Russian Federation)



Reactor type: Pressurized water reactor
Coolant/moderator: Light water
System pressure: 16.5 MPa
System temperature: 330°C
Thermal capacity: 20 MW(th)
Electrical capacity: 2.5 MW(e)
Design life: 25 years
Fuel material: UO_2–ZrO_2 (CERMET)
Fuel enrichment: 19.75%
Fuel cycle: 25 years
Design status: Conceptual design
Distinguishing features: CERMET fuel utilization, no core refuelling during lifetime

The UNITHERM concept is based upon NIKIET's experience in the design of marine nuclear installations.

The UNITHERM fuel element is designed as a cylindrical rod with four spacing ribs on its outer surface. The fuel is in the form of tiny blocks of UO_2 grains coated with zirconium and dispersed in a zirconium matrix. The gap between the fuel containing matrix and the cladding is filled with silumin. A fuel element of such design has a high uranium content and radiation resistance. These features taken together make it possible to operate such fuel elements during the whole specified core lifetime. The reactor core consists of 265 FAs installed in the plates of the removable reactor screen at the points of a regular hexagonal lattice.

A specific feature of the UNITHERM fuel cycle is the long and uninterrupted irradiation of fuel inside the reactor core throughout the whole reactor lifetime, with whole core refuelling. The metal ceramic (CERMET) fuel chosen for the UNITHERM is composed of UO_2 particles in a metallic (silumin or zirconium) matrix. This design is characterized by a high volume ratio of nuclear fuel; the use of the metallic matrix ensures minimum swelling and high thermal conductivity. An optimally shaped cladding is formed when the cladding is filled with the matrix composition.

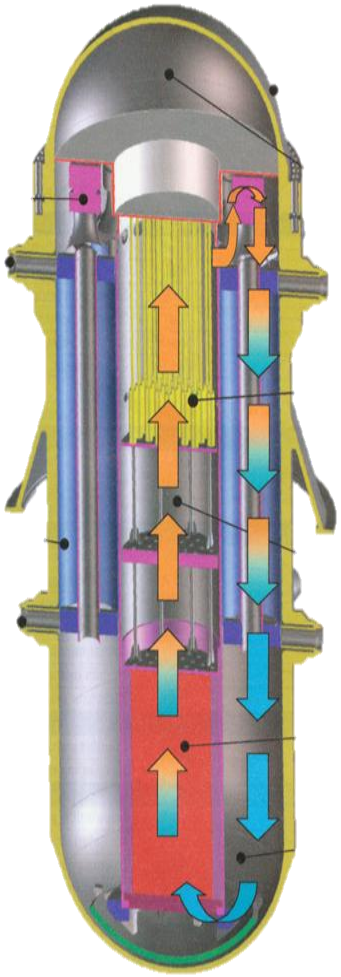
The UNITHERM design makes extensive use of passive systems and devices based on natural processes without external energy supply. These systems include:

- The control element drive mechanisms (CEDMs), designed to provide secure insertion of rods in the core by gravity;
- Locking devices in the CEDM to avoid unauthorized withdrawal of control rods;
- An independent passive heat removal system acting as a cooldown system in emergency shutdown of the reactor;
- A containment capable of maintaining primary coolant circulation as well as providing reactor cooldown and retention of radioactive products under the loss of primary circuit leaktightness;
- Passive systems for heat removal from the containment and biological shielding tanks.

The choice of a candidate turbine generator plant for the UNITHERM NPP depends on the plant capacity and operation mode requested by its users.

The UNITHERM NPP requires no major R&D for technology development. The detailed design stage would include qualification of the core, heat exchangers, CEDMs and other components [4].

IRIS (IRIS, International Consortium)



Full name: International reactor innovative and secure

Reactor type: Integral pressurized water reactor

Coolant/moderator: Light water

System pressure: 15.5 MPa

System temperature: 330°C

Thermal capacity: 1000 MW(th)

Electrical capacity: 335 MW(e)

Design life: 60 years

Fuel material: UO_2/MOX

Fuel enrichment: 4.95%

Fuel cycle: 48 months

Design status: Preliminary design completed, large scale testing for licensing process under way

Distinguishing features: Helical coil SGs, pressurizer, spool pumps and CRDMs fully internal to RPV; primary vessel without embrittlement; economies of scale within integral PWR concepts

IRIS is an LWR with a modular, integral primary system configuration. The concept is being pursued by an international group of organizations. IRIS is designed to satisfy four requirements: enhanced safety, improved economics, proliferation resistance and waste minimization. Its main features are:

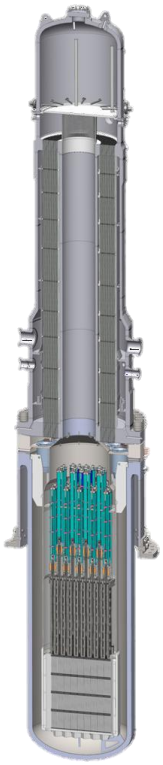
- Medium power of up to 335 MW(e) per module;
- Simplified compact design where the primary vessel houses the SGs, pressurizer and pumps;
- An effective safety approach;
- Optimized maintenance with intervals of at least four years.

The IRIS core is an evolutionary design based on conventional UO_2 fuel enriched to 4.95%. This fuel can be fabricated in existing facilities and is licensable to current requirements. FAs are constructed in a 17 × 17 lattice. The core contains 89 assemblies, each with an active fuel height of 4.27 m. Refuelling intervals of up to four years are possible. IRIS is designed to accommodate, without modification, a variety of core designs. Future core designs will include higher enriched UO_2 fuel and the capability to use mixed oxide (MOX) fuel. In the MOX case, IRIS is an effective actinide burner.

IRIS adopts passive safety systems and the safety by design philosophy including the risk informed approach. Due to IRIS's integral configuration, by design (i.e. with no intervention of either active or passive systems) a variety of accidents either are eliminated or their consequences and/or probability of occurring are greatly reduced. In fact, 88% of class IV accidents (the ones with the possibility for radiation release) are either eliminated outright or downgraded. This provides a high level defence in depth which may allow IRIS to claim no need for an emergency response zone.

The IRIS team has completed the preliminary design of the large scale test facility to prepare for the future design certification.

mPower (Babcock and Wilcox, USA)



Full name: B&W mPower reactor

Reactor type: Integral pressurized water reactor

Coolant/moderator: Light water

System pressure: 14.1 MPa

System temperature: 320°C

Thermal capacity: 500 MW(th)

Electrical capacity: 150 MW(e)

Fuel material: UO_2

Fuel enrichment: 5%

Fuel cycle: 4 years

Design status: Construction permit application to be submitted in mid–2012; design certification application to be submitted in the 4th quarter of 2013

Distinguishing features: Internal once through SG, pressurizer and control rod drive mechanism

The B&W mPower™ reactor module is an integral PWR designed by Babcock and Wilcox (B&W) to generate a power of 150 MW(e). The reactor design is a modular system in which the nuclear core, SGs, CRDM, coolant pumps and pressurizer are contained within a single vessel. The design offers flexibility so that multiple reactor modules can be aggregated to support local customer requirements and infrastructure constraints. The plant is designed to have the capacity to match customer demand in 150 MW(e) increments with a four year operating cycle without refuelling, using standard PWR fuel. The design life of the reactor is 60 years.

The reactor core consists of 69 FAs which have less than 5% enrichment, Gd_2O_3 spiked rods, Ag-Cd-In (AIC) and B_4C control rods, and a 3% shutdown margin. There is no soluble boron present in the reactor coolant for reactivity control. The FAs are of a conventional 17 × 17 design with a fixed grid structural cage. They have been shortened and optimized specifically for the mPower reactor.

The inherent safety features of the reactor design include a low core linear heat rate which reduces fuel and cladding temperatures during accidents, a large reactor coolant system volume which allows more time for safety system responses in the event of an accident, and small penetrations at high elevations increasing the amount of coolant available to mitigate a small break LOCA. The ECCS removes heat from the reactor core after anticipated transients in a passive manner while also passively reducing containment pressure and temperature. The plant is designed without taking credit for safety related emergency diesel generators, and a design objective is no core uncovery during design basis accidents.

A large pipe break LOCA is not possible because the primary components are located inside the pressure vessel and the maximum diameter of the connected piping is less than 7.6 cm.

In addition, the reactor features passive safety systems, a secure underground metal containment, sufficient to limit internal pressure for all design basis accidents, and on-site spent fuel storage capacity for the life of the reactor.

The balance of plant design consists of a conventional power train using a steam cycle and an optional air or water cooled condenser.

B&W and Bechtel Power Corporation entered into a formal alliance called Generation mPower to design, license and deploy mPower modular plants. A letter of intent has been signed with the Tennessee Valley Authority for joint development and pursuit of a construction permit and operating licence for up to six B&W mPower reactors [5].

NuScale (NuScale Power Inc., USA)

Reactor type: Integral pressurized water reactor

Coolant/moderator: Light water

System pressure: 12.8 MPa

Thermal capacity: 165 MW(th)

Electrical capacity: 45 MW(e)

Design life: 60 years

Fuel material: UO_2

Fuel enrichment: 4.95%

Fuel cycle: 24 months

Design status: NuScale plans to apply for design certification with the US Nuclear Regulatory Commission in 2012

Distinguishing features: Synergy through plant simplicity, reliance on existing light water technology and availability of an integral test facility

A NuScale plant can consist of 1–12 independent modules, each capable of producing a net electric power of 45 MW(e). Each module includes a pressurized LWR operated under natural circulation primary flow conditions. Each reactor is housed within its own high pressure containment vessel, which is submerged underwater in a stainless steel lined concrete pool.

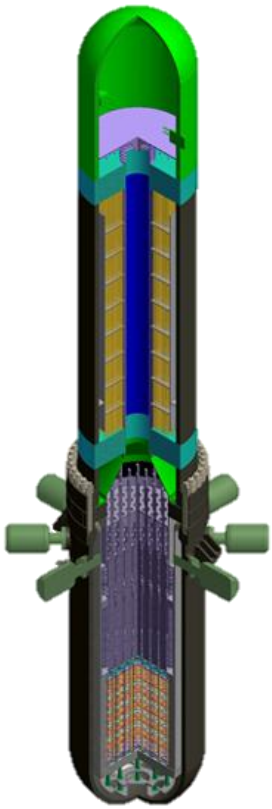
According to the design organization, there are five essential features of the NuScale plant that, in combination, distinguish it from the many other small nuclear plants being developed today:

- Compact size;
- Core cooled entirely by natural circulation;
- Reliance on well established LWR technology;
- Supported by a one third scale, electrically heated integral test facility which operates at full pressure and temperature;
- Utilization of a compact movable modular containment, in contrast to a traditional, cast in-place concrete design.

The NuScale plant includes a comprehensive set of engineered safety features designed to provide stable long term nuclear core cooling, as well as severe accident mitigation. They include a high pressure containment vessel, two passive decay heat removal and containment heat removal systems, a shutdown accumulator and severe accident mitigation. The multi-module NuScale plant spent fuel pool consists of a steel lined concrete pool located underground. It is designed with the capability of storing and cooling all of the fuel offloaded from 12 modules, as well as an additional 10 years' worth of used nuclear fuel. After 10 years of storage in the spent fuel pool, air cooled interim storage is possible, as with conventional nuclear plants. In 2003, Oregon State University, in collaboration with the Idaho National Engineering Laboratory and Nexant–Bechtel, completed a project to develop a preliminary design for an innovative reactor called the Multi-Application Small Light Water Reactor (MASLWR). In 2007, NuScale Power Inc. was formed to commercialize the concept and MASLWR was renamed the NuScale plant to reflect the significant improvements made to the original design. In early 2008, NuScale Power Inc. notified the US NRC of its intention to begin pre-application discussions aimed at submitting an application for design certification of a 12 module NuScale power plant.

NuScale Power Inc. plans to submit design for certification by the US NRC. This effort is aimed at deploying NuScale plants to address the near term needs of the US utility market.

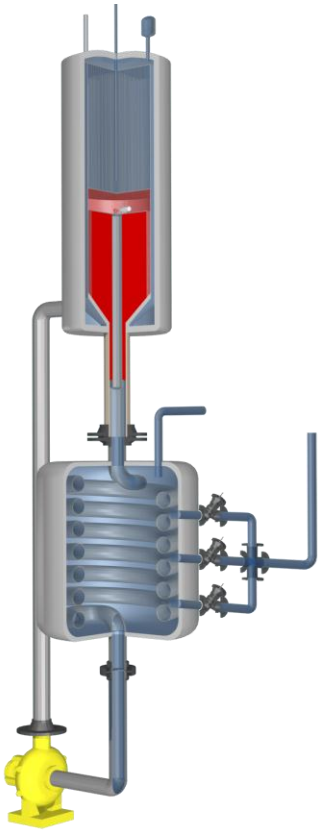
Westinghouse SMR (Westinghouse, USA)



Full name: Westinghouse small modular reactor
Reactor type: Integral pressurized water reactor
Coolant/moderator: Light water
System pressure: 15.5 MPa
System temperature: 310°C
Thermal capacity: 800 MW(th)
Electrical capacity: 225 MW(e)
Design life: 60 years
Fuel material: UO_2
Fuel enrichment: <5%
Fuel cycle: 24 months
Design status: Conceptual design
Distinguishing features: Incorporates passive safety systems and proven components of the AP 1000

The Westinghouse SMR is a 200 MW(e) class, integral PWR with all primary components located inside the reactor vessel.
The reactor core is a partial height version of the 17 × 17 fuel assembly design used in the AP1000 reactor. The active core height is 2.4 m with 89 FAs and 37 internal CRDMs. The fuel is 5% enriched and the core is designed to be refuelled every 24 months.
The CRDM utilizes latch assemblies, interfaces with fuel and controls that are based on existing designs. It also uses a three coil magnetic jack based on the AP1000, and a test programme has been initiated for the CRDM.
The reactor coolant pumps are three phase, horizontally mounted, axial flow pumps, each providing 0.8 m^3/s at 30.5 m of head. There are eight pumps, and they have a seal-less configuration that eliminates the need for pump seal injection. Each motor is rated at about 260 kW. The SGs are recirculating, once through straight tubes that achieve a compact physical envelope inside the reactor vessel. The hot leg of the reactor drives the coolant up through the middle of the tube bundle; the flow comes back down the tube bundle, from which the feedwater is circulated and sent to the external steam drum followed by the turbine.
The reactor vessel internals are based on the AP1000 design but modified to smaller dimensions. The pressurizer is integrated into the vessel head, thus minimizing the vessel size to 3.5 m in diameter and 24.7 m in height.
The safety features of the AP1000 are extended to the SMR design, and Westinghouse claims that no operator intervention is required for seven days following an incident. The containment vessel utilizes fully modular construction due to its small size of 9.8 m in diameter and 27 m in height.
The main components and facilities of the reactor plant are located below grade, including the containment, the spent fuel pool, the radioactive waste storage areas and the defence in depth facilities.
In February 2011, Westinghouse officially introduced the SMR design and is currently preparing for a role in the US Department of Energy's demonstration programme.
The design certification document is planned to be submitted to the US NRC in 2012.

FBNR (FURGS, Brazil)



Full name: Fixed bed nuclear reactor
Reactor type: Integral pressurized water
Coolant/moderator: Light water
System pressure: 16 MPa
System temperature: 326°C
Thermal capacity: 134 MW(th)
Electrical capacity: 72 MW(e)
Fuel material: UO_2
Fuel enrichment: 5%
Fuel cycle: 25 months
Design status: Conceptual description
Distinguishing features: CERMET fuels

The Fixed Bed Nuclear Reactor (FBNR) is a small reactor of 70 MW(e) with no need for on-site refuelling. It is a pressurized LWR with spherical fuel elements. It is simple in design, employing inherent safety features, passive cooling for some situations and proliferation resistant features, and it has a reduced environmental impact.

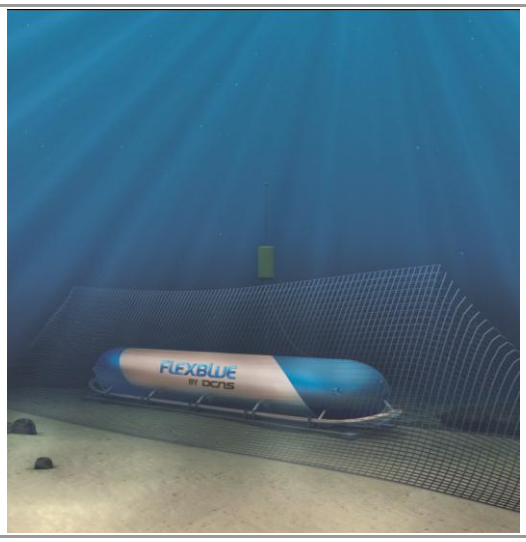
The reactor includes a pressurizer system and an integrated shell and tube SG in the upper portion, and the fuel chamber and reserve fuel chamber below. The spherical fuel elements are fixed in the suspended core by the flow of water coolant. Any accident signal will cut off the power to the coolant pump, causing a stop in the flow. This would make the fuel elements fall out of the reactor core, driven by gravity; these would enter the passively cooled reserve fuel chamber, where they would reside in a subcritical condition.

The reactor is modular in design, and each module is assumed to be fuelled at the factory. The fuelled modules in sealed form are then transported to the site. The FBNR has a long fuel cycle and, therefore, there is no need for on-site refuelling. Rather, the spent fuel containing module is shipped to an off-site spent fuel management facility.

The distinguishing feature of the FBNR design in terms of non-proliferation is that, under shutdown conditions, all fuel elements remain in the fuel chamber where only a single flange needs to be sealed and controlled for safeguards purposes. The spent fuel of the FBNR is in a convenient form and size that can be directly used as a source of radiation for irradiation purposes. A variety of irradiators can easily be constructed for applications in industry, agriculture and medicine.

The FBNR concept is being developed at the Federal University of Rio Grande do Sul (FURGS, Brazil) in cooperation with several international research groups. Relatively little work has been done for the FBNR so far, but the experience gained from the development of a fluidized bed reactor may facilitate the development of the FBNR [4].

Flexblue (DCNS, France)

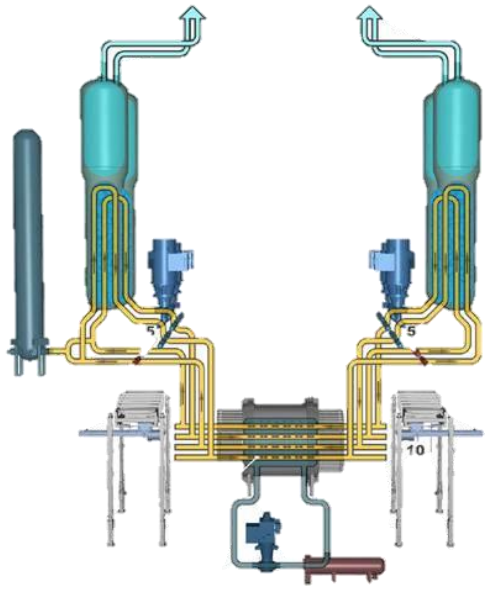


Reactor type: Subsea pressurized water reactor
Coolant/moderator: Light water
System pressure: 15.5 MPa
System temperature: 310°C
Thermal capacity: 150–750 MW(th)
Electrical capacity: 50–250 MW(e)
Design status: Conceptual description
Distinguishing features: Seabed, torpedo resistant, offshore power source

In France, the DCNS Company — designer of the reactors for French nuclear submarines — in partnership with AREVA-TA and CEA, is developing a small modular underwater NPP based on the state of the art French water cooled marine propulsion reactor. The whole plant, named Flexblue, is factory fabricated and fuelled and is transported to an operational (underwater) site by a surface ship. The reactor vessel appears as a torpedo resistant 100 m long cylinder weighing about 12 000 t. The electric output is between 50 and 250 MW(e) for a single NPP module, which is anchored on a flat seabed at a depth of between 60 and 100 m, several kilometres from the shore. The plant is remotely controlled and connected to the onshore grid by an underwater seabed embedded cable.

HEAVY WATER REACTORS

EC6 (AECL, Canada)



Full name: Enhanced CANDU 6

Reactor type: Pressure tube type reactor

Coolant/moderator: Heavy water (D_2O)

System pressure: 10 MPa

System temperature: 310°C

Thermal capacity: 2084 MW(th)

Electrical capacity: 740 MW(e)

Design life: 60 years

Fuel material: UO_2

Fuel enrichment: Natural uranium

Fuel cycle: Closed

Design status: Basic design

Distinguishing features: Improved CANDU 6 design

The Enhanced CANDU 6 (EC6) is a 740 MW(e) pressure tube reactor designed by Atomic Energy of Canada Limited (AECL). The design evolved from the CANDU 6 design.

The EC6 reactor assembly comprises the calandria vessel and end shields, 380 fuel channel assemblies, the reactivity control units and the calandria vault. The pressurized heavy water moderator is circulated through the calandria vessel, and the heavy water coolant flows through the fuel channel assemblies housed in the calandria vessel.

The fuel bundle consists of 37 elements. Each element contains three basic components: the UO_2 pellets, the sheath with canlub (graphite) coating in the inside surface and the end caps. The designers claim that various fuel cycle options can be accommodated in the EC6, including slightly enriched uranium up to 1.2%, thorium, MOX fuel and natural uranium equivalent fuel, which can be produced from recycled uranium from commercial NPPs. The design also incorporates on-line refuelling.

The major components of the heat transport system are the 380 reactor fuel channels with associated corrosion resistant feeders, four vertical SGs, four motor driven pumps, four reactor inlet headers, four reactor outlet headers, one electrically heated pressurizer and all necessary interconnecting piping and valves. The system is arranged in a two loop, figure eight configuration. The headers, SGs and pumps are all located above the reactor.

There are five safety systems broken into two groups in the EC6. Group 1 comprises the normal process and control systems, shutdown system 1 and the ECCS, with plant control and monitoring from the main control room. Group 2 comprises shutdown system 2, the emergency heat removal system (EHRS) and containment system, with plant monitoring and control of essential safety functions from the secondary control area. Group 2 also comprises the seismically qualified systems required to mitigate a design basis earthquake and the water injection and recovery systems for mitigation of severe accidents.

The turbine generator, feedwater and condensate plant are completely located within the turbine building and are part of the balance of plant. They are based on conventional designs and meet the design requirements specified by the nuclear steam plant designer to ensure the performance and integrity of the nuclear steam plant.

The Canadian Nuclear Safety Commission (CNSC) is currently conducting the pre-project design review of the EC6. This review identifies and mitigates any project risks due to licensing before project commitment. The CNSC's review includes two phases. Phase 1 involves a high level review of the safety design as a whole. Phase 2 deals with a complete review of design details and analysis.

PHWR-220 (NPCIL, India)

Full name: Pressurized heavy water reactor 220 MW(e)

Reactor type: Pressurized heavy water reactor

Coolant/moderator: Heavy water (D_20)

System pressure: 8.5 MPa

System temperature: 293°C

Thermal capacity: 755 MW(th)

Electrical capacity: 236 MW(e)

Design life: 40 years

Fuel material: UO_2

Fuel enrichment: Natural uranium

Fuel cycle: 24 months

Design status: Operational

Distinguishing features: Proven design, indigenous Indian effort

The Indian pressurized heavy water reactor (PHWR) programme consists of 220 MW(e), 540 MW(e) and 700 MW(e) units. At present, India is operating sixteen 220 MW(e) units at five atomic power stations.

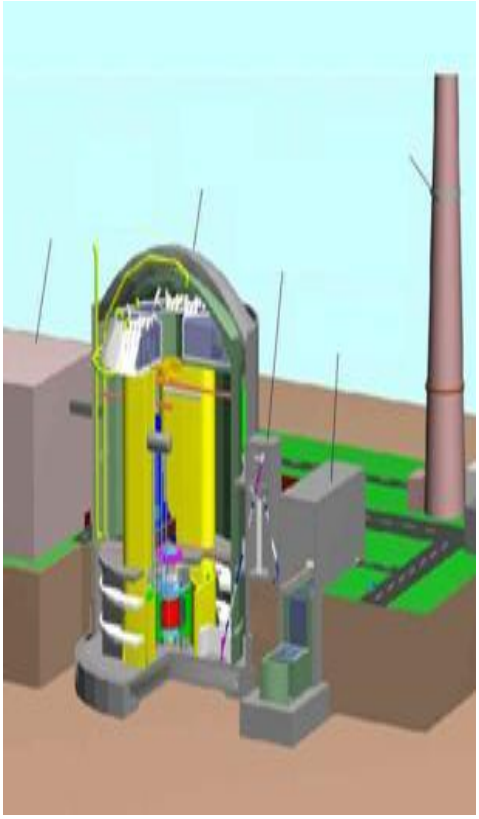
The PHWR uses heavy water as the moderator and coolant and natural uranium dioxide as the fuel. The reactor consists of an integral assembly of two end shields and a calandria, with the latter being submerged in the water filled vault.

PHWRs use natural uranium in dioxide form as fuel. During the residence period in the reactor, about 1% of the uranium is burned. Increase in fuel burnup beyond 15 000 MW·d/t U using higher fissile content materials like slightly enriched uranium, MOX and thorium oxide in place of natural uranium in fuel elements used in 220 MW(e) PHWRs is being studied. Due to their higher fissile content these bundles will be capable of delivering higher burnup than the natural uranium bundles. The maximum burnup studied with these bundles is 30 000 MW·d/t U.

Indian PHWRs are designed and operated to achieve the fundamental safety objectives in conformity with regulatory requirements of codes, guides and standards. The licensing process is well established, with a multi-tier review carried out by the Nuclear Power Corporation of India Limited (NPCIL) and the regulatory body. The well established principle and practice of defence in depth is followed. The reactor regulating system is used for normal power manoeuvring, including fast reduction of power as a setback action. Reactor shutdown is achieved by two diverse and fast acting shutdown systems supplemented by a slow acting poison injection system for maintaining long term subcriticality. The shutdown systems are designed so that the first shutdown system is the preferred mode of shutdown.

The Indian PHWR design facilitates the effective implementation of safeguards. The provisions made for this purpose include installation of equipment (such as cameras, bundle counters, core discharge monitors, etc.), an IAEA equipment control room, reliable power supply and lighting arrangements in the vicinity of cameras. Spent nuclear fuel inventory is reduced due to an increase in fuel burnup under normal operation. The above activities reduce spent fuel discharge and hence reduce quantities of spent nuclear fuel.

AHWR300-LEU (BARC, India)



Full name: Advanced heavy water reactor with low enriched uranium and thorium mixed oxide fuel

Reactor type: Pressure tube type heavy water moderated reactor

Coolant/moderator: Light water/heavy water (D_2O)

System pressure: 7 MPa

System temperature: 285°C

Design life: 100 years

Thermal capacity: 920 MW(th)

Electrical capacity: 304 MW(e)

Fuel material:(Th, ^{233}U)–MOX and (Th, Pu)–MOX

Fuel enrichment: 3.00–3.75% ^{233}U/2.5–4.0% Pu

Fuel cycle: Closed

Design status: Basic design

Distinguishing features: Mixed oxide thorium closed fuel cycle; vertical cooling channels

The Indian Advanced Heavy Water Reactor with Low Enriched Uranium and Thorium Mixed Oxide Fuel (AHWR300-LEU) was designed and developed by the Bhabha Atomic Research Centre (BARC) to achieve large scale use of thorium for the generation of commercial nuclear power. This reactor is designed to produce most of its power from thorium, with no external input of ^{233}U in the equilibrium cycle.

The AHWR300-LEU design utilizes a calandria vessel housing the core. The calandria vessel is filled with heavy water as the moderator and has vertical cooling channels with boiling light water as the primary coolant. The AHWR300-LEU circular fuel cluster is designed with 30 (Th, ^{233}U)–MOX pins and 24 (Th, Pu)–MOX pins; 452 of the fuel clusters comprise the full core. The AHWR300-LEU is also designed to use a closed fuel cycle, recovering ^{233}U and thorium from the spent fuel to be used in the manufacture of fresh fuel.

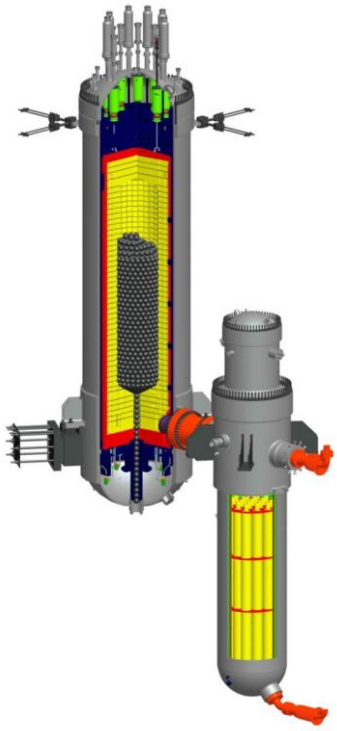
The coolant circulation is driven by natural convection through tail pipes to steam drums, where steam is separated for running the turbine cycle. During shutdown, passive valves establish communication of steam drums with the isolation condensers submerged inside an 8000 m^3 gravity driven water pool for decay heat removal under hot shutdown conditions.

The reactor design's safety features include a variety of passive safety systems such as the injection of emergency core coolant through rupture disks, containment isolation following a large break LOCA and passive poison injection by use of system steam pressure in the event of active shutdown system failure. The ECCS is designed to provide core cooling for 72 h following an incident.

The primary function of the steam and feed system is to transfer heat produced in the reactor core to the turbine for production of electrical power. The steam and feed system forms an interface between the main heat transport system and the ultimate heat sink (seawater) and provides the means for heat removal at various reactor operating conditions. The steam and feed system consists of the steam mains, turbo generator and auxiliaries, condensing system, condensate and feedwater heating system, steam dumping and relief systems and on-line full condensate flow purification. Construction of the AHWR300-LEU is likely to commence in 2011. Evaluation of probable sites is in progress.

GAS COOLED REACTORS

HTR-PM (Tsinghua University, China)



Full name: High temperature gas cooled reactor — pebble bed module
Reactor type: High temperature gas cooled pebble bed reactor
Coolant/moderator: Helium/graphite
System pressure: 7 MPa
System temperature: 750°C
Thermal capacity: 500 MW(th)
Electrical capacity: 211 MW(e)
Design life: 40 years
Fuel material: UO_2, UC_2, UCO
Fuel enrichment: 8.9%
Fuel cycle: Open
Design status: Under construction
Distinguishing features: Prototype for commercial sized HTGR

In March 1992, the State Government approved the construction of the 10 MW pebble bed high temperature gas cooled test reactor (HTR-10). In January 2003, the reactor reached full power (10 MW). Tsinghua University's Institute of Nuclear Energy Technology (INET) has completed many experiments to verify crucial inherent safety features of modular high temperature reactors including:

- Loss of off-site power without any countermeasures;
- Main helium blower shutdown without any countermeasures;
- Loss of main heat sink without any countermeasures;
- Withdrawal of all rods without any countermeasures;
- Helium blower trip without closing outlet cut-off valve.

The second step of HTGR application in China began in 2001 [4, 6] when the HTR–PM project was launched.

The HTR–PM utilizes the triple coated isotropic (TRISO) ceramic coated particle fuel element, which contains fuel kernels of 200–600 µm UO_2, UC_2 and UCO, but can also contain thorium or plutonium. The various layers of the TRISO fuel element enable it to tolerate more than 1600°C and retain fission products.

The primary circuit consists of the RPV, the SG pressure vessel and the hot gas duct vessel connecting the former. The core is a ceramic cylindrical shell housing the pebble bed, which acts as a reflector, heat insulator and neutron shield.

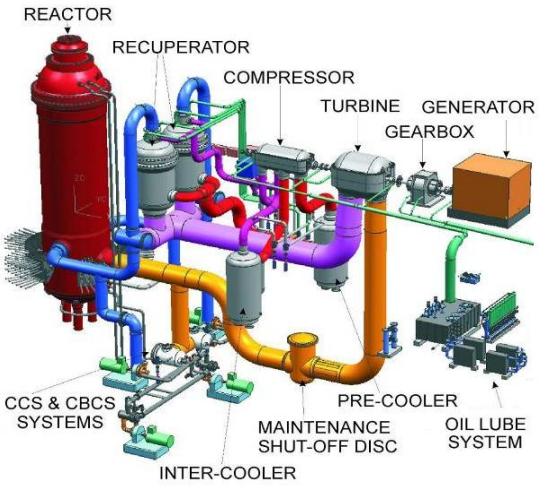
The SG is a vertical, counterflow, once through generator with a helium–water interface. There are multiple units consisting of helical heat transfer tubes.

The HTR–PM incorporates the inherent safety principles of the modular high temperature gas cooled reactor (MHTGR), which removes the decay heat passively from the core under any designed accident conditions and keeps the maximum fuel temperature below 1600°C so as to contain nearly all fission products inside the SiC layer of the TRISO coated fuel particles. This eliminates the possibility of core melt and large releases of radioactivity into the environment.

There is no ECCS present in the design, and the decay heat is removed by natural mechanisms such as heat conduction or heat radiation. The reactor cavity cooling system operates using cooling panels connected to an air cooler, but the decay heat can be sufficiently removed if this system is not operating.

In 2004, the HTR–PM standard design was started jointly by INET and CHINERGY. In 2006, the project was listed in the national guidelines on medium and long term programmes for science and technology development, and the Huaneng Shandong Shidaowan Nuclear Power Co., Ltd, the owner of the HTR–PM, was established by the China Huaneng Group, the Nuclear Industry Construction Group and Tsinghua University.

PBMR (PBMR Pty, South Africa)



Full name: Pebble bed modular reactor

Reactor type: High temperature gas cooled pebble bed reactor

Coolant/moderator: Helium/graphite

System pressure: 9 MPa

System temperature: 900°C

Design life: 40 years

Thermal capacity: 400 MW(th)

Electrical capacity: 165 MW(e)

Fuel material: UO_2 in coated particles

Fuel enrichment: 9.6%

Fuel Residence Time: 31 months

Design status: Basic design completed

Distinguishing features: Tall slender core with annular fuel cavity and fixed central column reflector coupled to a direct closed gas turbine cycle

The pebble bed modular reactor (PBMR) is an HTGR based on German HTR module designs. Its design has the following attributes, which the designer expects will contribute to enhanced safety:

- Use of TRISO fuel particles shown to remain intact to at least 1600˚C, with delayed failure fractions at even higher temperatures;
- A geometry that allows the passive dissipation of decay heat to an external heat sink;
- Relatively low power density to aid in the limitation of fuel temperatures following a loss of coolant due to a leak that cannot be isolated;
- Load following limited to 100–40–100% to reduce the excess operating reactivity to a value that prevents fuel failure for any scenario of group control rod withdrawal without scram;
- Use of helium as the coolant, which avoids the effects of phase changes and has a negligible effect on plant reactivity when pressures fluctuate;
- Control rods move only in the reflector and thus avoid any problem of mechanical damage to the fuel spheres;
- Optimized heavy metal fuel loading (water ingress is virtually eliminated in the direct gas cycle power conversion design), which cannot cause undue reactivity addition;
- Use of nuclear grade graphite to ensure minimal corrosion by impurities and low activation at end of life.

Plant performance targets are based on 93% power operation for a lifetime of 40 years. In addition, provision is made to replace the inner part of the reflectors once every 20 years.

Based on these assumptions, the following targets have been adopted:

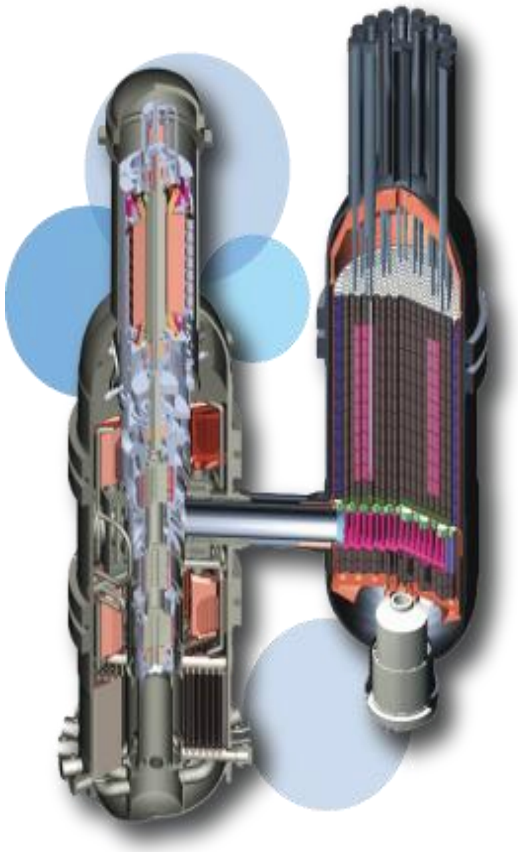
- Reliability: One unplanned outage per year;
- Maintenance intervals of 5–6 years for helium turbine maintenance;
- Construction time of 36 months for the Nth plant.

The short construction time is based on completed site preparation work, modular factory construction of all major systems and concurrent construction wherever possible.

Fuel costs are highly dependent on quantity, as the coated particle and sphere fabrication are major cost items during operations and maintenance.

The project to build a demonstration unit and the associated fuel plant were abandoned in 2010.

GT–MHR (General Atomics, USA)



Full name: Gas turbine modular helium reactor

Reactor type: High temperature gas cooled reactor

Coolant/moderator: Helium/graphite

System pressure: 6.39 MPa

System temperature: 750°C

Thermal capacity: 350 MW(th)

Electrical capacity: 150 MW(e)

Design life: 60 years

Fuel material: UCO

Fuel enrichment: 15.5%

Fuel cycle: 18 months

Design status: Conceptual design

Distinguishing features: Efficient production of hydrogen by high temperature electrolysis or thermochemical water splitting

The gas turbine modular helium reactor (GT–MHR) couples an HTGR with a Brayton power conversion cycle to produce electricity at high efficiency. Because of its capability to produce high coolant outlet temperatures, the modular helium reactor system can also efficiently produce hydrogen by high temperature electrolysis or thermochemical water splitting.

The GT–MHR direct Brayton cycle power conversion system contains a gas turbine, an electric generator and gas compressors. The use of the direct Brayton cycle results in a net plant efficiency of approximately 48%.

The GT–MHR gas turbine power conversion system has been made possible by utilizing large active magnetic bearings, compact, highly effective gas to gas heat exchangers and high strength, high temperature steel alloy vessels.

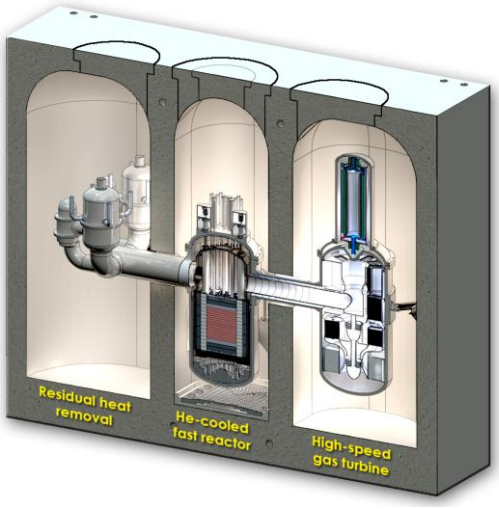
The standard fuel cycle for the commercial GT–MHR utilizes low enriched uranium (LEU) in a once through mode without reprocessing. General Atomics claims that the GT–MHR produces less heavy metal radioactive waste per unit energy produced because of the plant's high thermal efficiency, high fuel burnup and lower fertile fuel inventory. Similarly, the GT–MHR produces less total plutonium and ^{239}Pu (materials of proliferation concern) per unit of energy produced.

The GT–MHR safety design objective is to provide the capability to reject core decay heat relying only on passive (natural) means of heat transfer without the use of any active safety systems. The GT–MHR safety concept is centred on retention of the radionuclides in the fuel under all normal and postulated accident conditions to the degree that doses at the site boundary will be within the US Environmental Protection Agency's radionuclide protective action guidelines, without reliance on AC powered systems or operator action.

The GT–MHR fuel form presents formidable challenges to diversion of materials for weapons production, as either fresh or as spent fuel.

Pre-application licensing interactions with the NRC began in 2001 including submission of a licensing plan. From a technology development standpoint, the path forward for deployment of the GT–MHR technology is necessarily a demonstration project, such as the next generation nuclear plant project [1].

EM^2 (General Atomics, USA)

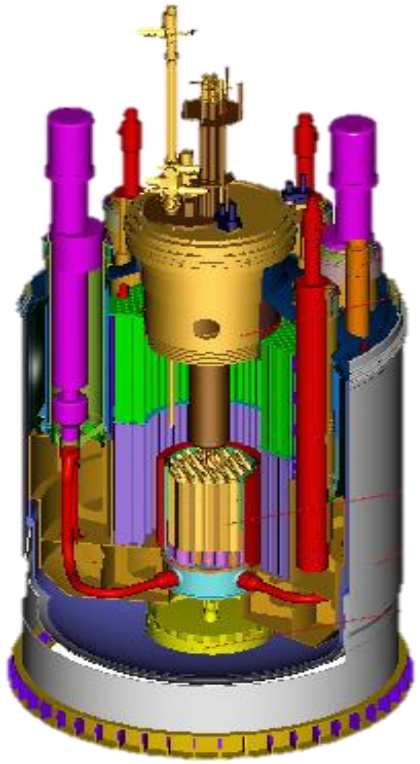


Full name: Energy Multiplier Module
Designer: General Atomic
Reactor type: High temperature gas cooled fast reactor
Coolant/moderator: Helium
System temperature: 850°C
Thermal capacity: 500 MW(th)
Electrical capacity: 240 MW(e)
Design life: 30 years
Fuel material: Used nuclear fuel
Fuel enrichment: 1% ^{235}U, 1% Pu, MA
Fuel cycle: 30 years
Design status: Conceptual design
Distinguishing features: Helium cooled fast reactor; reduces spent fuel inventories

The EM^2 is designed as a modification of an earlier high temperature, helium cooled reactor. It is an effort to utilize used nuclear fuel without conventional reprocessing.
The reactor is designed to produce 500 MW(th) and 240 MW(e) based on a closed cycle gas turbine. The EM^2 is a fast reactor design intended to burn used nuclear fuel and has a 30 year core without the need for refuelling or reshuffling. The spent fuel cladding is first removed and the fuel is pulverized and processed using the atomics international reduction oxidation (AIROX) dry process to remove fission products. The fuel burned in the reactor is recycled upon discharge.
In a first generation plant, the fuel consists of about 22.2 t of LEU starter and about 20.4 t of used nuclear fuel. The used nuclear fuel is roughly 1% ^{235}U, 1% Pu and mixed actinides (MA), and 3% fission products; the rest is ^{238}U. The design organization claims that there is no need for uranium enrichment after the first generation reactor, as the discharge from the preceding generation is used for the succeeding generation. Out of each discharge, about 38.5 t is used in the succeeding generation while about 4 t of fission products is removed.
Using a gas turbine cycle, the designers claim to achieve 48% efficiency with a core outlet temperature of 850°C. The entire containment is designed to be below grade and sealed for the 30 year core period [7].

LIQUID METAL COOLED REACTORS

CEFR (CNEIC, China)



Full name: Chinese experimental fast reactor

Reactor type: Liquid metal cooled fast reactor

Coolant: Sodium

System temperature: 530°C

Thermal capacity: 65 MW(th)

Electrical capacity: 20 MW(e)

Design life: 30 years

Fuel material: $(Pu,U)–O_2$

Fuel enrichment: 19.6% ^{235}U

Design status: Operating

Distinguishing features: Provides fast neutrons for irradiation testing, test bed for actual fast reactor conditions, model verification and experience needed for a commercial reactor

Based on two phases of fast reactor R&D targeting a 60 MW(th) experimental fast reactor, the China experimental fast reactor (CEFR) project was launched by the China Nuclear Energy Industry Corporation (CNEIC) in the framework of the national high tech programme. The main objective of the CEFR is to accumulate experience in fast reactor design, fabrication of components, construction, pre-operational testing, and operation and maintenance.

The CEFR is a sodium cooled, 65 MW(th) experimental fast reactor with $PuO_2–UO_2$ fuel, but with UO_2 as the first loading. The reactor core is composed of 81 FAs, three compensation subassemblies and two regulation subassemblies. Three safety subassemblies act as the secondary shutdown system. A total of 336 stainless steel reflector subassemblies and 230 shielding subassemblies, in addition to 56 positions for primary storage of spent fuel subassemblies, are included.

The reactor core and its support structure are supported on lower internal structures. Two main pumps and four intermediate heat exchangers (IHXs) are supported on upper internal structures.

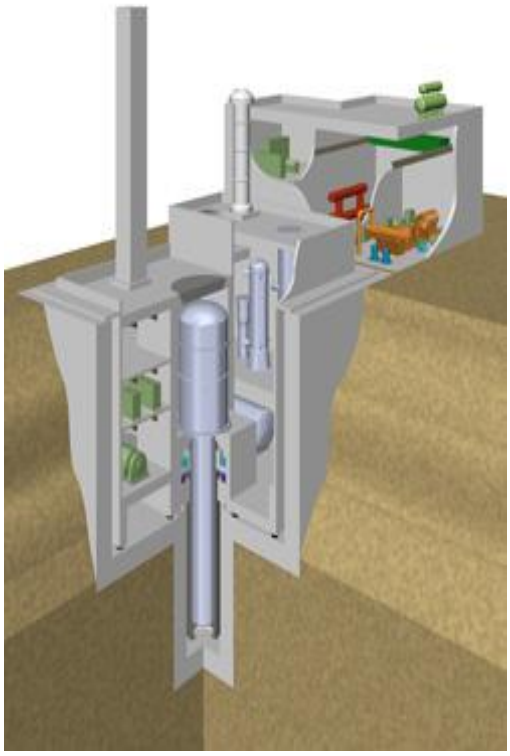
The primary circuit is composed of main pumps, four IHXs, reactor core support diagrid plenum, pipes and cold and hot sodium pools. In normal operation the average sodium temperature in the cold pool is 360°C, and in the hot pool it is 516°C. The secondary circuit has two loops; each one is equipped with one secondary pump, two IHXs, an evaporator, a super heater, an expansion tank and valves. The outlet sodium temperature of the secondary circuit from the IHX is 495°C. When the sodium leaves the evaporator it will decrease to 310°C, and the outlet of the super heater is 463.3°C. The tertiary water steam circuit provides 480°C and 14 MPa superheated steam to the turbine.

The use of liquid sodium coolant provides a low pressure system that is insensitive to postulated coolant boundary failures because coolant flashing cannot occur. Sodium is contained within the reactor vessel, along with the core, the primary pumps and the IHXs, except the purification system. The CEFR is a small reactor that has greater heat inertia than many other pool reactors due to its larger primary sodium loading per MW(th). This provides long time margins for corrective action in the event of a loss of heat sink.

Two independent passive decay heat removal systems are designed for the CEFR. Each one is rated to a thermal power of 0.525 MW(th) under working conditions; the decay heat is removed by natural convection and circulation of primary and secondary coolant and natural draft by air.

The CEFR achieved first criticality in July 2010, began procedures for physical startup in August 2010, and is currently connected to the grid operating at 40% capacity [8].

4S (Toshiba, Japan)



Full name: Super safe small and simple

Reactor type: Liquid metal cooled fast reactor

Coolant: Sodium

System temperature: 510°C

Thermal capacity: 30 MW(th)

Electrical capacity: 10 MW(e)

Design life: 30 years

Fuel material: U–Zr Alloy

Fuel cycle: 30 years

Design status: Working with the city of Galena, AK, USA, for potential application there

Distinguishing features: For the initial fuel load, plutonium will be extracted from the spent fuel of existing light water reactors

The 4S is a sodium cooled reactor without on-site refuelling. Being developed as a distributed energy source for multipurpose applications, the 4S offers two outputs: 30 MW(th) and 135 MW(th). The 4S is not a breeder reactor, since the blanket fuel, usually consisting of depleted uranium located around the core to absorb leakage neutrons from the core to achieve breeding of fissile materials, is not present in its basic design.

The 4S is a reactor whose core has a lifetime of approximately thirty years. The movable reflector surrounding the core gradually moves, compensating the burnup reactivity loss over the thirty year lifetime. The reactor power can be controlled by the water–steam system without affecting the core operation directly, which makes the reactor applicable for a load follow operation mode.

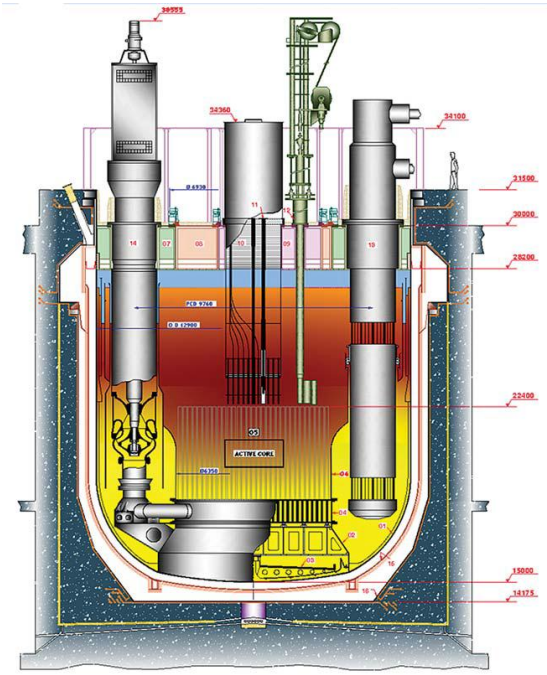
The reactor is an integral pool type, as all primary components are installed inside the reactor vessel. Major primary components are IHXs, primary electromagnetic pumps, moveable reflectors which form a primary reactivity control system, the ultimate shutdown rod, radial shielding assemblies, the core support plate, coolant inlet modules and fuel subassemblies.

To reduce the probability of component failure, the design eliminates active systems and feedback control systems from the reactor side as well as components with rotating parts. There is also limitation of the radioactivity confinement area, since there is no refuelling during the life of the reactor. Other objectives include: the prevention of core damage in accidents, the confinement of radioactive materials, and the prevention of sodium leakage and the mitigation of associated impacts if leakage should occur.

Technical features of the 4S contributing to a high level of proliferation resistance include the use of uranium based fresh fuel with ^{235}U enrichment less than 20% by weight and a low plutonium content in the spent fuel (less than 5% by weight). The reprocessing technology available for metal (alloy) fuel, such as U–Zr or U–Pu–Zr, ensures that plutonium is always recovered together with the accompanying minor actinides, which include highly radioactive and radiotoxic nuclides.

The 4S design is being developed at Toshiba and the Central Research Institute of the Electric Power Industry (CRIEPI) in Japan; Chubu Electric Power Company supported the initial phase of R&D relevant to the 4S. R&D focusing on core, fuel and reflector technologies was conducted under the sponsorship of the Ministry of Education, Culture, Sports, Science and Technology (MEXT) in Japan, which included CRIEPI, the Japan Atomic Energy Research Institute (JAERI), Osaka University and the University of Tokyo. Pre-application review by the US NRC was commenced in 2007 [4, 8].

PFBR-500 (IGCAR, India)



Full name: Prototype fast breeder reactor 500 MW(e)

Reactor type: Fast breeder reactor

Coolant: Sodium

System temperature: 547°C

Thermal capacity: 1250 MW(th)

Electrical capacity: 500 MW(e)

Design life: 40 years

Fuel material: PuO_2–UO_2

Fuel cycle: 6 months

Design status: Under construction

Distinguishing features: First industrial scale fast breeder reactor in India

A 500 MW(e) prototype fast breeder reactor (PFBR) is being built by the Indira Gandhi Centre for Atomic Research (IGCAR) at Kalpakkam; it will be first reactor of its kind in India. The PFBR will be a 500 MW(e) (1250 MW(th)), two loop, sodium cooled, pool type reactor. It will utilize MOX fuel and depleted UO_2 as a blanket.

The reactor core is made up of 1758 subassemblies (SAs), arranged in a hexagonal lattice. Of these, 181 fuel SAs form the active core. There are two enrichment regions in the active core for power flattening. There are two rows of radial blanket SAs and 12 absorber rods, comprising 9 control and safety rods and 3 diverse safety rods arranged in two rings. Enriched boron carbide is used as the absorber material. The radial core shielding is provided by stainless steel and B_4C SAs. They limit the secondary sodium activity and radiation damage and activation of the primary circuit components to acceptable levels.

The PFBR is designed to have a vertical configuration, and offload refuelling is envisaged for it. It is designed to require refuelling after every 185 effective full power days of the reactor. In one refuelling campaign, 62 fuel SAs, 25 blanket SAs and 5 absorber SAs will be replaced.

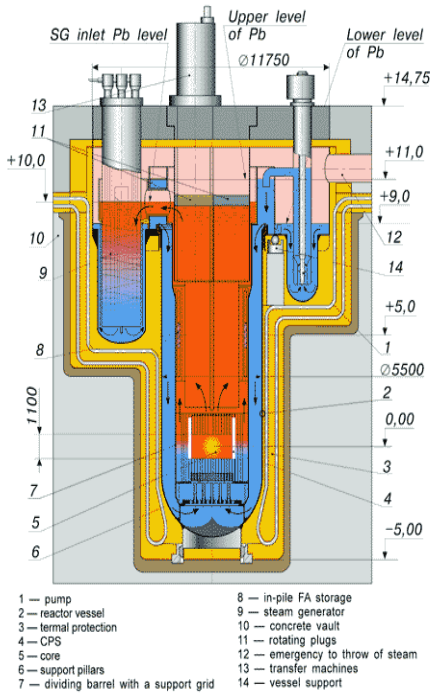
The core is surrounded by fertile blankets and in-vessel shielding. The IHX and sodium pumps are in the pool. The in-vessel shield is provided to reduce radiation damage to the inner vessel, secondary sodium activation, activation of the IHX and sodium pumps, and axial leakages through the bottom fission product gas plenum. Mockup shielding experiments were carried out to optimize design of this in-vessel shield.

There are eight integrated SG units, four per secondary loop where steam at 763 K and 17.2 MPa is produced. Four separate safety grade decay heat exchangers are provided to remove the decay heat directly from the hot pool. The hot and cold pool sodium temperatures are 820 K and 670 K, respectively.

There are four sodium to sodium decay heat exchangers dipped in the hot pool to remove decay heat when the normal heat removal path is not available. These are shell and tube type heat exchangers similar to the IHX. The entire primary sodium including core and inner vessel is housed in a large vessel known as the main vessel (MV). A safety vessel is provided around the MV in the unlikely event of a sodium leak in the MV. There are two independent secondary sodium circuits. Each secondary sodium circuit consists of a pump, two IHXs, one surge tank and four SGs. Except for the SG, the steam and water system is similar to that of a conventional thermal power plant.

The PFBR is currently under construction and is expected to be commissioned in 2012 [9].

BREST-OD-300 (RDIPE, Russian Federation)



BREST–300 reactor. Vertical section

Full name: Fast reactor natural safety

Reactor type: Liquid metal cooled fast reactor

Coolant: Lead

System temperature: 540°C

Thermal capacity: 700 MW(th)

Electrical capacity: 300 MW(e)

Fuel material: PuN–UN

Fuel cycle: 10 months

Design status: Preliminary design

Distinguishing features: High level of inherent safety due to natural properties of the fuel, core and cooling design

BREST is a lead cooled fast reactor fuelled with uranium plutonium mononitride (PuN–UN) that uses a two circuit heat transport system to deliver heat to a supercritical steam turbine.

The adopted fuel exhibits high density and high conductivity, and is compatible with lead and the fuel cladding of chromium ferritic martensitic steel. To provide a significant coolant flow area, the level of power removed by natural lead circulation is increased, the coolant preheating temperature is reduced and cooling losses in the damaged FA are primarily excluded in the case of local flow rate blockage, no core FAs have shrouds. The FA design allows radial coolant overflow in the core which prevents overheating of the damaged FA.

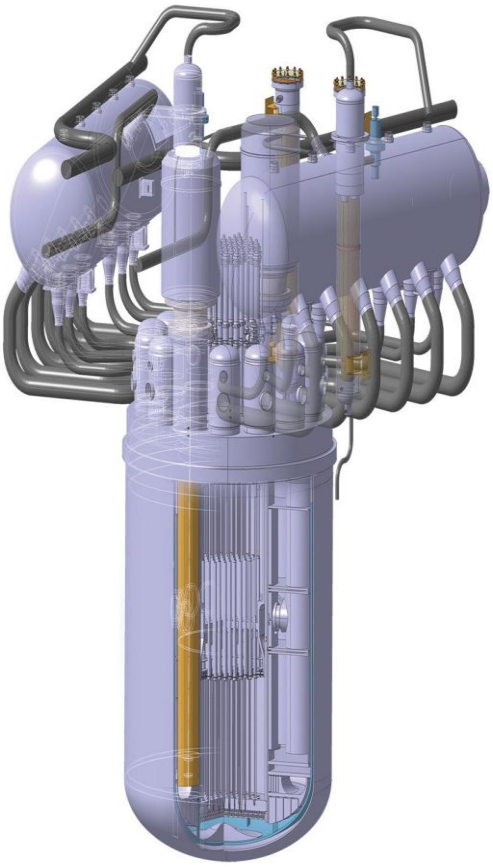
BREST-OD-300 uses a mixed integral loop configuration of the primary circuit, as the SG and main coolant pumps are installed outside the reactor vessel. The reactor and the SGs are located in the thermally shielded concrete vault, without using a metal vessel. The concrete temperature is kept below the allowable limit by means of natural air circulation.

Accidents are avoided thanks to the intrinsic safety features of BREST, including the reactivity fuel temperature coefficient, coolant and core design components, and coolant pressure and temperature at the core inlet and outlet. The safety analysis has shown that none of the considered initial events involving a fast introduction of reactivity up to its full margin, interruption of the forced coolant circulation, loss of secondary heat sink or lead super cooling at the core inlet leads to accidents with fuel damage and inadmissible radioactive or toxic releases, even in the case of a failure of the reactor's active safety systems.

The secondary circuit is a non-radioactive circuit consisting of SGs, main steam lines, a feedwater system and one turbine unit with supercritical steam parameters. A standard K–300–240–3 turbine unit with a three cylinder (HPC+MPC+LPC) steam condensation turbine with intermediate steam superheating is used.

The BREST-OD-300 power unit is designed as a pilot and demonstration unit intended for studying the reactor facility operation in different modes and optimizing all processes and systems that support the reactor operation. Furthermore, BREST-OD-300 is also considered the prototype of a fleet of medium sized power reactors. Indeed, after the operational tests, the unit will be commissioned for electricity supply to the grid [8].

SVBR-100 (AKME Engineering, Russian Federation)



Full name: Lead–bismuth fast reactor

Reactor type: Liquid metal cooled fast reactor

Coolant: Lead–bismuth

System temperature: 500°C

Thermal capacity: 280 MW(th)

Electrical capacity: 101 MW(e)

Design life: 60 years

Fuel material: UO_2

Fuel enrichment: 16.3%

Fuel cycle: 7–8 years

Design status: Detailed design

Distinguishing features: Closed nuclear fuel cycle with mixed oxide uranium plutonium fuel, operation in a fuel self-sufficient mode

The SVBR-100 is an innovative small modular fast reactor with lead–bismuth coolant (LBC). In the Russian Federation, lead–bismuth cooled reactor technology has been used in eight different nuclear submarines. The experience gained from these reactors included: ensuring the corrosion resistance of structural materials, controlling the LBC quality and the mass transfer processes in the reactor circuit, ensuring the radiation safety of the personnel carrying out work with equipment contaminated with the ^{210}Po radionuclide, and multiple LBC freezing and unfreezing in the reactor facility.

The SVBR-100's reactor core operates without any partial refuelling. The fresh fuel is loaded as a single cartridge while the spent nuclear fuel is unloaded cassette by cassette. The core configuration allows for a lower power density compared with the nuclear submarines using LBC reactors.

This design has the capability to utilize various fuel cycles. The first stage will be the typical uranium oxide fuel leading to a core breeding ratio (CBR) of 0.84; MOX fuel can also be used, leading to a CBR just below 1. Using UO_2 as the starting fuel, the closed fuel cycle can be realized in 15 years. Nitride uranium and uranium plutonium fuel can also be used to improve safety and fuel cycle characteristics.

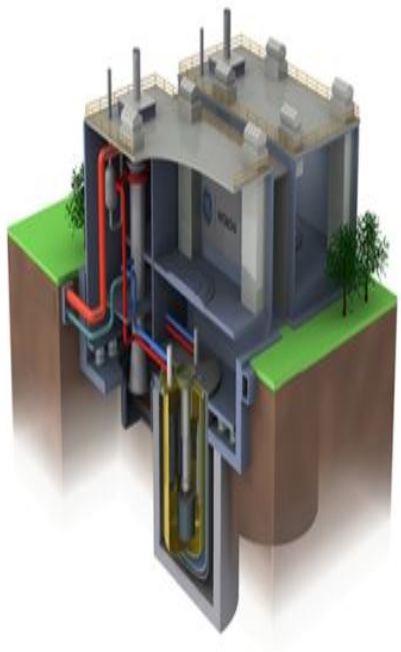
The SVBR-100 design pursues a high level of safety with inherent self-protection and passive safety by use of chemically inert LBC and integral arrangement of the primary circuit equipment in a single vessel operating at approximately atmospheric pressure. This allows the reactor design to exclude many safety systems required for traditional type reactors and to simplify and reduce the cost of the power plant.

Safety systems in the reactor plant include fusible locks of auxiliary safety rods to provide passive shutdown, bursting disk membrane to prevent over pressurization and passive removal of residual heat in the event of a blackout.

The SVBR-100 reactor pursues resistance to nuclear fissile material proliferation by using uranium with enrichment below 20% while using uranium oxide fuel in the initial core. The reactor is designed to operate for eight years without core refuelling.

The Rosatom Scientific and Technical Council held on 15 June 2006 approved the development of the technical design of experimental industrial power unit based on the SVBR-100 [4, 8, 10].

PRISM (GE-Hitachi, USA)



Full name: Power reactor innovative small module
Reactor type: Liquid metal cooled fast breeder reactor
Coolant: Sodium
System temperature: 485°C
Thermal capacity: 471 MW(th)
Electrical capacity: 155 MW(e)
Fuel material: U–Pu–Zr
Fuel enrichment: 26% Pu,10% Zr
Design status: Detailed design
Distinguishing features: Underground containment on seismic isolators with a passive air cooling ultimate heat sink; part of the advanced recycling centre for spent nuclear fuel

The PRISM design uses a modular, pool type, liquid sodium cooled reactor. The reactor fuel elements are a uranium–plutonium–zirconium metal alloy. The reactor uses passive shutdown and decay heat removal features.

The PRISM reactor core was designed to meet several objectives:

- To limit peak fuel burnup;
- To limit the burnup reactivity swing;
- To provide an 18 month refuelling interval;
- To provide a 54 month life for the fuel;
- To provide a 90 month life for the blankets.

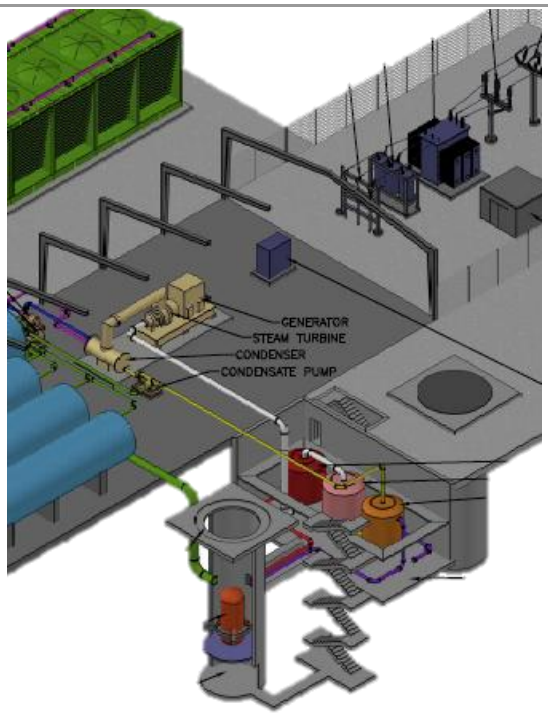
The reactor is designed to use a heterogeneous metal alloy core. The core consists of 42 FAs, 24 internal blanket assemblies, 33 radial blanket assemblies, 42 reflector assemblies, 48 radial shield assemblies and 6 control and shutdown assemblies.

The primary heat transport system is contained entirely within the reactor vessel. The flow path goes from the hot pool above the core through the IHXs, where it is cooled; the sodium exits the IHX at its base and enters the cold pool. The cold pool sodium is then drawn through the fixed shield assemblies into the pump inlet manifold. The four electromagnetic pumps take suction from the cold pool sodium through a manifold and discharge into the high pressure core inlet plenum through the piping connecting each manifold to the plenum. The sodium is then heated as it flows upward through the core and back into the hot pool.

The designers state that the inherent shutdown characteristics of the reactor core are a diverse and independent means of shutdown in addition to the control rod scram. The passive features are composed of several reactivity feedback properties including: the Doppler effect, sodium density and void, axial fuel expansion, radial expansion, bowing, control rod drive line expansion and reactor vessel expansion. The negative feedbacks maintain the reactor at a safe, stable state at an elevated temperature, but the reactor may still be critical if none of the reactor control rods have been inserted. The ultimate shutdown system has been added to bring the reactor to a subcritical state.

A PRISM power block consists of three reactor modules, each with one SG, that collectively supply one turbine generator set. Steam from the three SGs is combined and supplied at near saturation conditions to the high pressure inlet of the turbine generator. The exhaust steam enters the two low pressure turbine sections after it passes through moisture separators and reheaters. The steam is exhausted and, after passing through a series of condensers and coolers, enters the feedwater and condensate system [11, 12].

HPM (Hyperion Power Generation Inc., USA)



Full name: Hyperion power module
Reactor type: Liquid metal cooled reactor
Coolant: Lead–bismuth
System temperature: 500°C
Thermal capacity: 70 MW(th)
Electrical capacity: 25 MW(e)
Design life: 5–15 (nominal 10) years
Fuel material: Uranium nitride
Fuel enrichment: 19.75%
Fuel cycle: 10 years
Design status: Conceptual design
Distinguishing features: Transportable factory fuelled design

Hyperion Power Generation Inc. was formed to develop the Hyperion power reactor, first conceived at the Los Alamos National Laboratory (LANL) in New Mexico. Through the commercialization programme at LANL's Technology Transfer Division, Hyperion Power Generation was awarded the exclusive licence to utilize the intellectual property and develop the module.

The reactor has been designed to deliver 70 MW of power for a ten year lifetime, without refuelling. The materials in the core are uranium nitride fuel, HT–9 as the structural material, lead–bismuth eutectic (LBE) as the coolant, quartz as the radial reflector, and B4C rods and pellets for in-core reactivity control. The reactor is approximately 1.5 m in diameter and 2.5 m in height, in which there are 24 subassemblies containing the fuel pins. The pin assembly is filled with liquid LBE to provide a high conductivity thermal bond between the fuel and cladding. The gap in the fuel pins has been sized to preclude fuel clad mechanical interference throughout the core's lifetime. A plenum is located at one end, which serves as both a fission gas plenum and a repository for the LBE inside the pin as the fuel swells with burnup.

The core coolant is LBE, with a mixed mean exit temperature of 500°C. This temperature limits the cladding temperature so that maximum cladding creep over the 10 year lifetime of the reactor is less than 1%. During operational shutdown, decay heat is removed from the HPM by two methods. The first method transfers heat from the core by natural circulation of coolants in the primary and secondary loops to the SGs. The second removes heat by passive vaporization of water from the surface of the secondary containment vessel.

There are two independent, safety grade reactivity control systems in the core: a control rod system comprising 18 B_4C control rods and a reserve shutdown system consisting of a central cavity into which B_4C spheres may be inserted into the core. Both the control rods and the spheres are inserted into dry wells in the core, which are hexagonally shaped thimbles. These thimbles penetrate the reactor vessel and are sealed from the primary coolant. Both systems can independently take the core to long term cold shutdown.

The safety concept of the HPM is driven by a set of design criteria the designers believe are sufficient to ensure protection of the facility and its surroundings. These criteria are sealing the core, operational simplicity, minimal to no in-core moving, mechanical components and separation of power production and conversion operations.

According to the HPM conceptual baseline schedule, the HPM design will be finalized in March 2012 [13].

Appendix

SUMMARY OF SMR DESIGN STATUS

Reactor design	Reactor type	Designer, country	Capacity(MW(e))/ configuration	Design status
CNP-300	Pressurized water reactor	CNNC, China	325	In operation
PHWR-220	Pressurized heavy water reactor	NPCIL, India	236	In operation
CEFR	Liquid metal cooled fast reactor	CNEIC, China	20	In operation
KLT-40S	Pressurized water reactor	OKBM Afrikantov, Russian Federation	35 × 2 modules barge mounted	Under construction
HTR-PM	High temperature gas cooled pebble bed reactor	Tsinghua University, China	211	Under construction
PFBR-500	Liquid metal cooled fast breeder reactor	IGCAR, India	500	Under construction
EC6	Pressurized heavy water reactor	AECL, Canada	740	Detailed design; CANDU 6 reference plants are in operation
CAREM	Integral pressurized water reactor	CNEA, Argentina	27	Design certification to prepare for prototype deployment by 2012

SMART	Integral pressurized water reactor	KAERI, Republic of Korea	100	Detailed design
ABV-6M	Pressurized light water Reactor	OKBM Afrikantov, Russian Federation	8.6 × 2 modules, barge mounted land based	Detailed design
RITM-200	Integral pressurized water reactor	OKBM Afrikantov, Russian Federation	50	Detailed design
VBER-300	Pressurized water reactor	OKBM Afrikantov, Russian Federation	325	Detailed design
WWER-300	Pressurized water reactor	OKBM Gidropress, Russian Federation	300	Detailed design
IRIS	Integral pressurized water reactor	IRIS, International Consortium	335	Detailed design
mPower	Integral pressurized water reactor	B&W, USA	150 × 4 modules	Pre-application interactions with the US NRC in July 2009, design certification application expected to be submitted 4th quarter of year 2012
NuScale	Integral pressurized water reactor	NuScale Power Inc., USA	45 × 12 modules	NuScale plans to apply for design certification with the US NRC in 2012
SVBR-100	Liquid metal cooled fast reactor	AKME Engineering, Russian Federation	101	Detailed design

PRISM	Liquid metal cooled fast breeder reactor	GE-Hitachi, USA	155	Detailed design
4S	Liquid metal cooled reactor	Toshiba, Japan	10	Detailed design
IMR	Integrated modular water reactor	Mitsubishi Heavy Industries, Japan	350	Conceptual design
VK-300	Boiling water reactor	RDIPE, Russian Federation	250	Conceptual design
UNITHERM	Pressurized water reactor	RDIPE, Russian Federation	2.5	Conceptual design
Westinghouse SMR	Integral pressurized water reactor	Westinghouse, USA	225	Conceptual design
FBNR	Integral pressurized water reactor	FURGS, Brazil	72	Conceptual design
Flexblue	Subsea pressurized water reactor	DCNS, France	50–250 seabed anchored	Conceptual design
AHWR300-LEU	Pressure tube type heavy water moderated reactor	BARC, India	304	Basic design
PBMR	High temperature gas cooled pebble bed reactor	PBMR Pty, South Africa	165	Conceptual design
GT–MHR	High temperature gas cooled reactor	General Atomics, USA	150	Conceptual design
EM^2	High temperature gas cooled fast reactor	General Atomics, USA	240	Conceptual design

BREST-OD-300	Liquid metal cooled fast reactor	RDIPE, Russian Federation	300	Conceptual design
HPM	Liquid metal cooled fast reactor	Hyperion Power Generation Inc., USA	25 × N modules, Single module or multimodule	Conceptual design

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