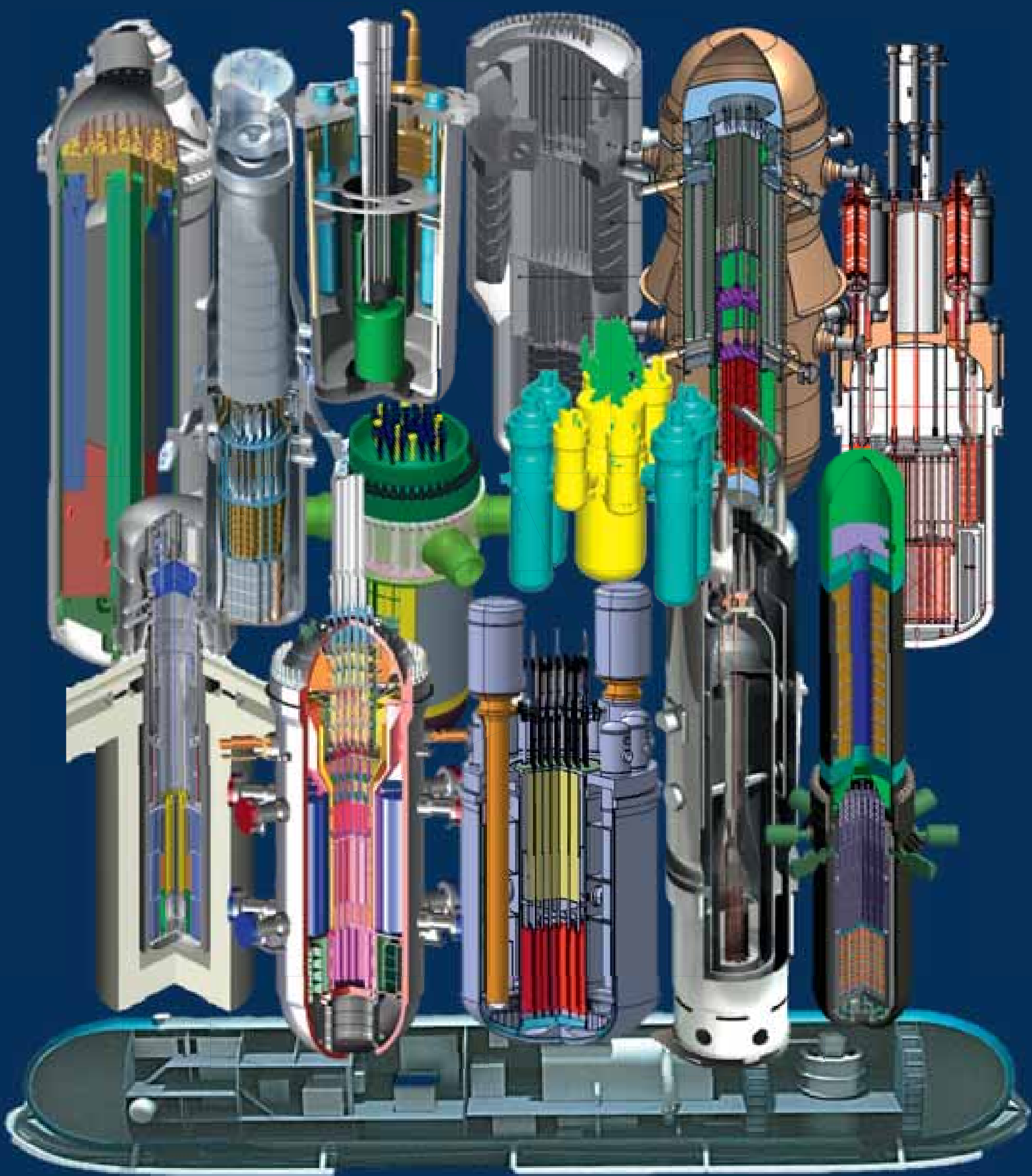
STATUS OF SMALL AND MEDIUM SIZED REACTOR DESIGNS

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

http://aris.iaea.org



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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 SMR 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 and 700 MW(e). Worldwide, 131 SMR units are in operation in 26 Member States, with a capacity of 59 GWe. At present, 14 SMRs are under construction in 6 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. SMRs are under development for all principal reactor lines: 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, Brazil, France, Japan, the Republic of Korea, the Russian Federation and the United States of America. 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. Site excavation work for a 27 MW(e) CAREM prototype was completed at the end of August 2012 and construction has begun. 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, which has a thermal capacity of 330 MW(th), is intended for sea water desalination and received standard design approval in 2012. In the Russian Federation, seven 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 LWR with natural circulation of the primary coolant, and is in the detailed design stage. The RITM-200, an integral reactor with forced circulation for universal nuclear icebreakers, is designed to provide 8.6 MW(e). The VK-300 is a 250 MW(e) simplified boiling water reactor (BWR) that operates with natural circulation and employs passive residual heat removal systems (RHRSs). 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. 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. The N.A. Dollezhal Research and Development Institute of Power Engineering (NIKIET) is designing the UNITHERM PWR, based on design experience in marine nuclear installations, and the SHELF PWR, a 6 MW(e) underwater, remotely operated power source.

In the USA, three integral pressurized water SMRs are under development: Babcock & Wilcox's mPower, NuScale and the Westinghouse SMR. The mPower design consists of two 180 MW(e) modules and its design certification application is expected to be submitted to the US Nuclear Regulatory Commission (NRC) in the fourth quarter of 2013. NuScale Power envisages a nuclear power plant made up of twelve 45 MW(e) modules and plans to apply for design certification to the NRC in 2013. 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. All three designs have submitted applications to the US Department of Energy for funding to support 'first of a kind' engineering, design certification and licensing [1]. 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 nuclear power plant with an output of 160 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 (CANDU) reactor series, which offers various power ratings. The Enhanced CANDU 6 (EC6) is an advanced 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 to 540 to 700 MW(e), are under construction or in operation. The 304 MW(e) advanced heavy water reactor (AHWR300-LEU) design incorporates vertical pressure tubes, low enriched uranium and thorium fuel, and passive safety features; it is currently in the basic design phase.

With regard to GCRs, several designs in the SMR classification are under development in China, South Africa and the USA. China has developed, constructed and operated the HTR-10, an experimental pebble bed helium cooled high temperature reactor (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 (GT-MHR) is a conceptual design that has the potential to produce hydrogen by high temperature electrolysis or thermochemical water splitting. Finally, the energy multiplier module (EM2) 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 China Experimental Fast Reactor (CEFR), a sodium cooled 20 MW(e) experimental fast reactor with PuO_2–UO_2 fuel, is currently in operation and was connected to the grid in 2011. India is building the 500 MW(e) Prototype Fast Breeder Reactor (PFBR), which is expected to be commissioned in 2013. 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 a design control document is currently being drafted to plan the licensing process. The Gen4 Module (G4M) design with an electrical power output of 25 MW(e) is in the conceptual design stage.

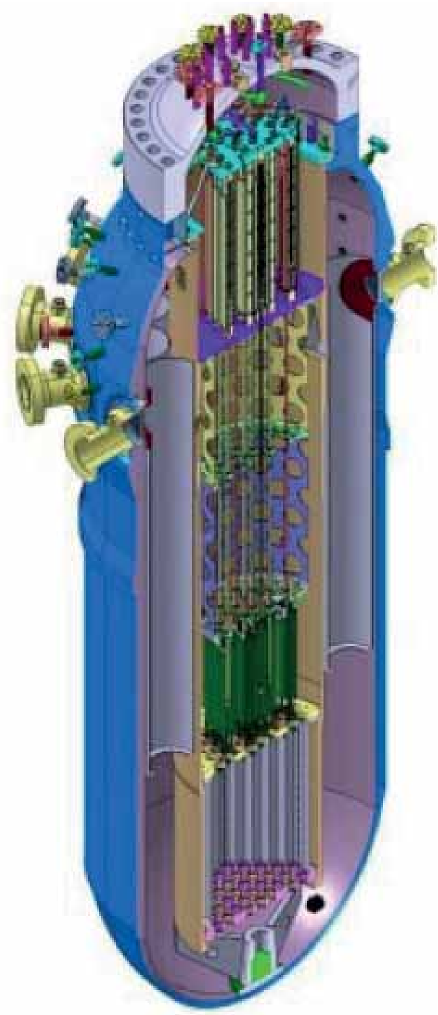
This booklet provides a two page description of the given SMR designs. The first page provides a brief technical summary with the main reactor parameters and the second page provides a description of the reactor and its various systems. The predictions of core damage frequencies (CDFs), seismic design and detailed plant and reactor parameters are provided by the design organizations without validation or verification by the IAEA.

LIGHT

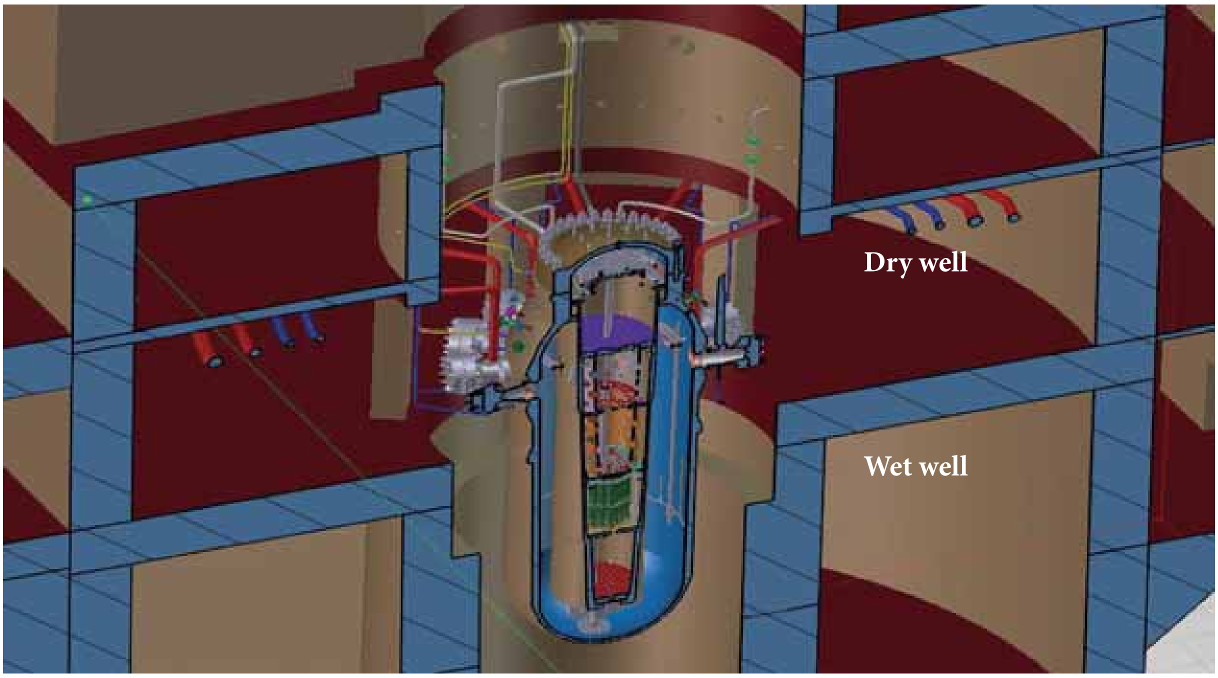
WATER

REACTORS

CAREM (CNEA, Argentina)



Reactor type:	Integral pressurized water reactor
Electrical capacity:	25 MW(e)
Thermal capacity:	100 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Natural circulation
System pressure:	12.25 MPa
Core outlet temperature:	326°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	UO_2
Fuel enrichment:	3.1%
Fuel cycle:	14 months
Reactivity control:	Rod insertion
No. of safety trains:	2
Emergency safety systems:	Passive
Residual heat removal systems:	Passive
Design life:	60 years
Design status:	Site preparation
Seismic design:	N/A
Predicted core damage frequency:	N/A
Planned deployment:	2016
Distinguishing features:	Pressure suppression containment

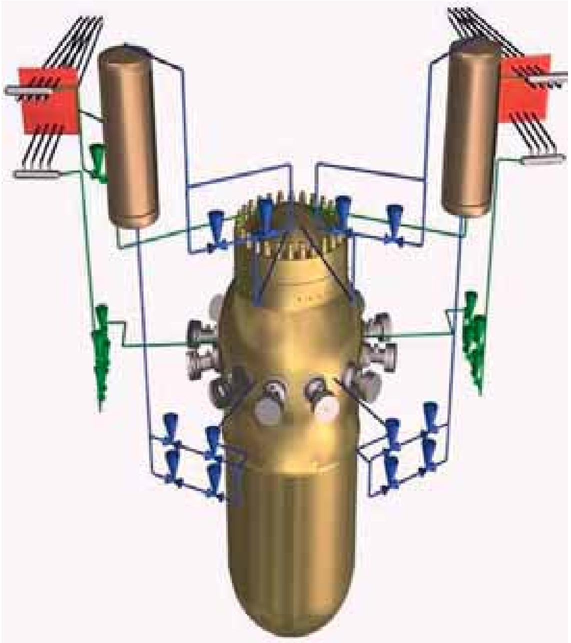


Pressure suppression containment.

Introduction

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

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.



Reactor pressure vessel with a safe shutdown system and a residual heat removal system.

Description of the nuclear systems

Due to the integrated design approach, the pressurizer and steam generators (SGs) are located inside the reactor pressure vessel. 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.

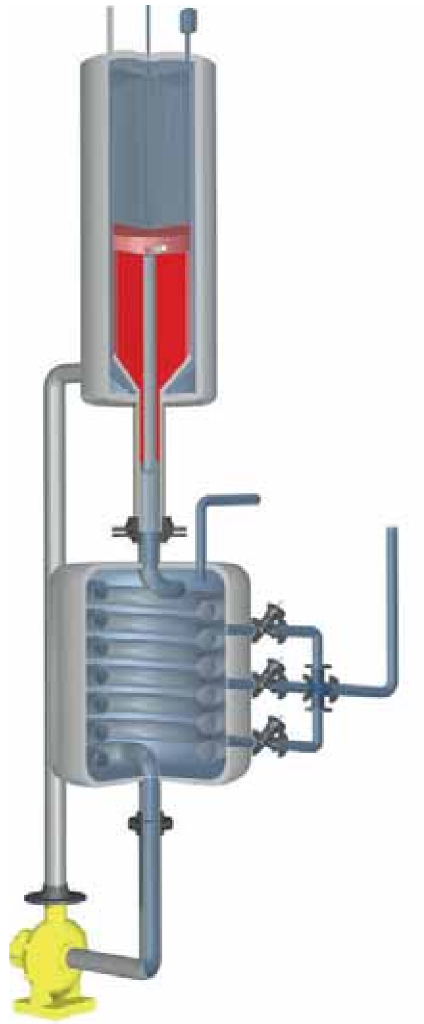
Description of the safety concept

The defence in depth concept has been internalized in the design, in order to improve safety significantly, compared with the current generation of nuclear power plants. Many intrinsic characteristics contribute to the preclusion of typical LWR initiating events, such as large and medium loss of coolant accidents, 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 h for each of the two redundancies). The safety systems are duplicated to fulfil the redundancy criteria, and the shutdown system is diversified to fulfil regulatory requirements.

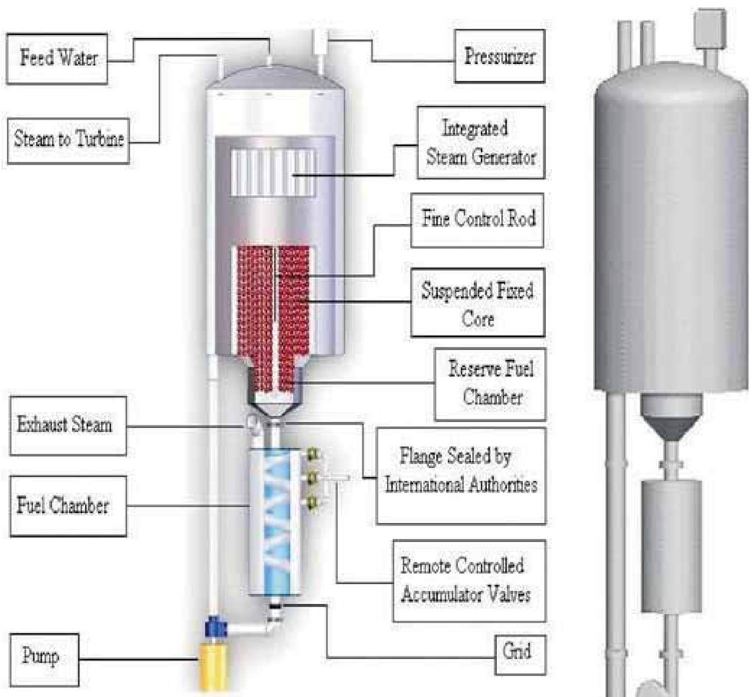
Deployment status and planned schedule

The CAREM concept was presented in 1984 in Lima, Peru, during the IAEA conference on SMRs [2]. CAREM was one of the first of the new generation reactor designs. Site excavation work began in 2011 and contracts and agreements between stakeholders are under discussion. The next step of the project is to start construction of the 100 MW(th) demonstration plant (CAREM-25) by 2012.

FBNR (FURGS, Brazil)



Reactor type:	Pressurized water reactor
Electrical capacity:	72.0 MW(e)
Thermal capacity:	218.4 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Forced
System pressure:	16 MPa
Core outlet temperature:	326°C
Thermodynamic cycle:	Indirect Rankine
Fuel material:	CERMET UO_2
Fuel enrichment:	5%
Fuel cycle:	25 months
Reactivity control:	Rod insertion and soluble boron
No. of safety trains:	N/A
Emergency safety systems:	Passive
Residual heat removal systems:	Passive
Design life:	N/A
Design status:	Concept description
Seismic design:	N/A
Predicted core damage frequency:	N/A
Planned deployment:	N/A
Distinguishing features:	Small reactor without on-site refuelling



Schematic Design of FBNR

Introduction

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 with a reduced environmental impact.

The reactor includes a pressurizer system and an integrated shell and tube steam generator in the upper portion, with 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.

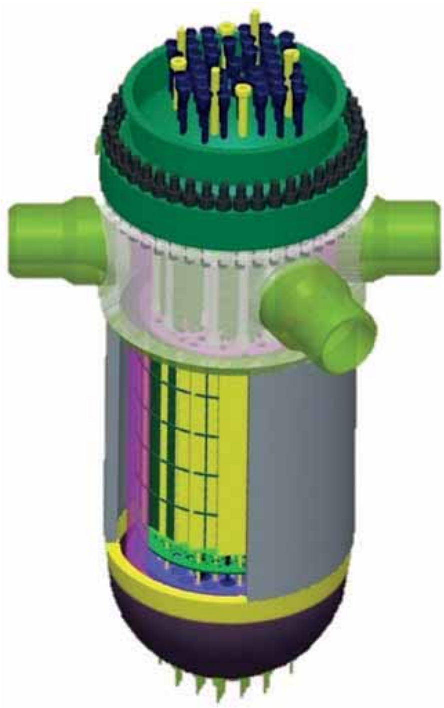
The reactor is modular in design, and each module is assumed to be fuelled at the factory. The fuelled modules in sealed form are then transported to the site. The FBNR has a long fuel cycle and, therefore, there is no need for on-site refuelling. Rather, the spent fuel containing module is shipped to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

Deployment status

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 [10].

CNP-300 (CNNC, China)



Reactor type:	Pressurized water reactor
Electrical capacity:	325 MW(e)
Thermal capacity:	999 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Forced circulation
System pressure:	15.2 MPa
Core outlet temperature:	302°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	UO_2
Fuel enrichment:	2.4–3.0%
Fuel cycle:	18 months
Reactivity control:	Soluble boron and rod insertion
No. of safety trains:	2
Emergency safety systems:	Active and passive systems
Residual heat removal systems:	Active systems
Design life:	40 years
Design status:	In operation
Seismic design:	0.25g
Predicted core damage frequency:	3.04E-4/reactor year
First date of completion:	15 Dec. 1991 Qinshan, China
Distinguishing features:	Operating small two loop PWR

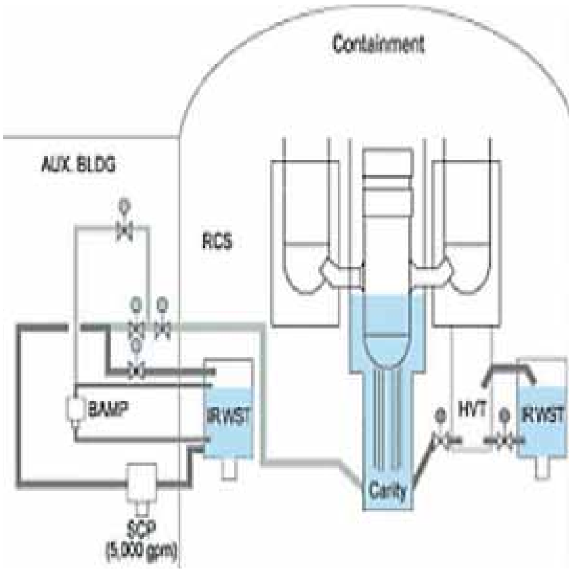
Introduction

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

The systems and the major equipment of the nuclear island, including the NSSS, are designed by the Shanghai Nuclear Engineering Research and Design Institute, and the system of the conventional island is designed by the East China Electric Power Design Institute.

Description of the nuclear systems

The reactor core consists of 121 fuel assemblies (FAs). Each FA is composed of uranium dioxide



Scheme for in-vessel retention.

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, a reactor coolant system (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 a steam generator (SG). The reactor coolant pumps are vertical, single stage, axial flow pumps. 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.

Description of the safety concept

Several engineered safety features have been incorporated into the plant design to reduce the consequences of a loss of coolant accident (LOCA). These safety features include a safety injection system, which 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 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 between the fuel building and the containment through a fuel transfer tube.

Description of the turbine-generator systems

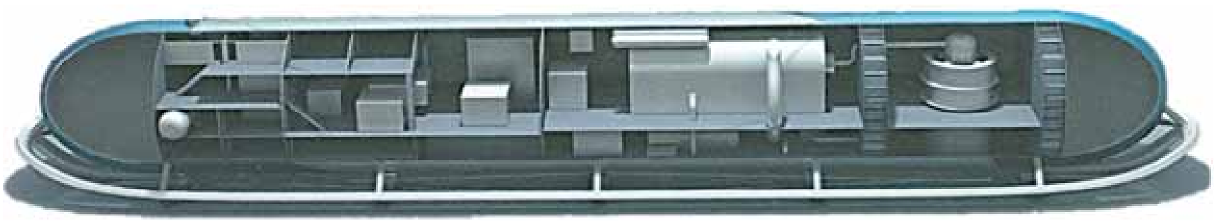
The electric power system connections to the plant are designed to provide a diversity of reliable power sources which are physically and electrically isolated from each other. The turbine-generator is a tandem compound, comprising three cylinders and four flow units with intermediate moisture separator-reheaters. The turbine is equipped with a digital electrohydraulic control system having the advantages of a controller based on microprocessors and a high pressure, fire resistant fluid supply, used for control of turbine valve operation.

Deployment status and planned schedule

The first CNP-300 unit, Qinshan-1, was connected to the grid in 1991 and two CNP-300 units have been exported to Pakistan, Chashma units 1 and 2 [3].

Flexblue (DCNS, France)

Reactor type:	Pressurized water reactor
Electrical capacity:	160 MW(e)
Thermal capacity:	600 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Forced
System pressure:	15.5 MPa
Core outlet temperature:	310°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	UO2
Fuel enrichment:	5%
Fuel cycle:	36 months
Reactivity control:	Soluble boron
No. of safety trains:	N/A
Emergency safety systems:	Passive
Residual heat removal systems:	Passive
Design life:	60 years
Design status:	Conceptual design
Seismic design:	N/A
Predicted core damage frequency:	N/A
Planned deployment:	N/A
Distinguishing features:	Transportable nuclear power plant; submerged operation



Plant layout.

Introduction

Flexblue is a 160 MW(e) transportable and submarine nuclear power unit, operating up to a depth of 100 m. It is 140 m long.

Description of the nuclear systems

The power production cycle lasts 3 years. At the end of a production cycle, the unit is taken back to its support facility. The reactor core is then refuelled and periodic maintenance is carried out. A major overhaul is scheduled every 10 years.

At the end of its life, the power unit is transported back to a dismantling facility, which results in a quick, easy and full recovery of the natural site.

Reactivity is controlled without soluble boron. This allows a simplified primary chemistry management, with reduced radioactive effluent and waste to the environment.

Ship transportable units.

Description of the safety concept

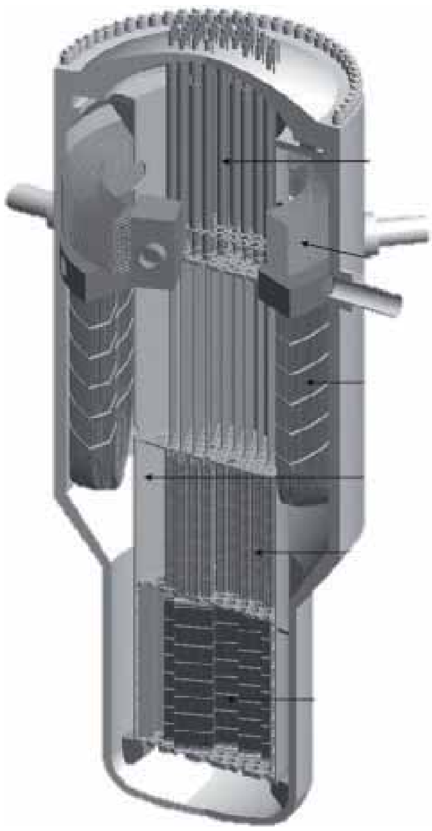
DCNS claims that Flexblue offers an extended level of nuclear safety that is enhanced by the submarine environment and based exclusively on proven technologies from the nuclear, naval and offshore industries. Water offers a natural protection against most of the possible external hazards and guarantees a permanently and indefinitely available heat sink.

In this framework, the use of passive safety systems brings the reactor to a safe and stable state without external intervention for an indefinite period of time.

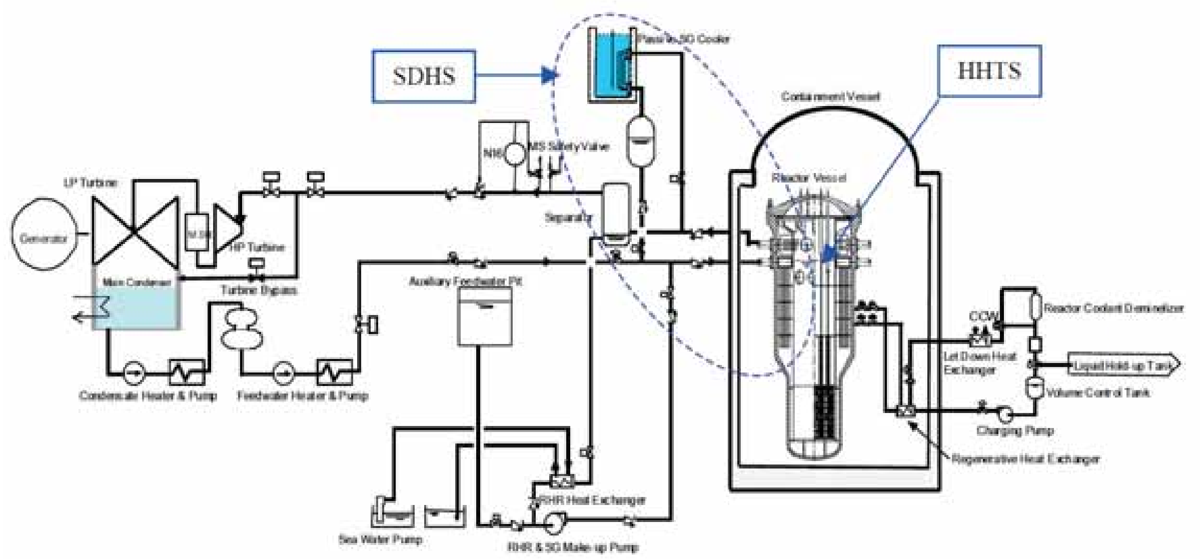
Deployment capability

The unit is transported on a carrier ship, moored on the seafloor and remotely operated from a coastal command and control room. During exploitation, maintenance teams gain regular access to the module with an underwater vehicle. This power unit can be used within a 'nuclear farm' that includes other modules. This allows the owner to increment the number of units that are operated on the site, depending on the power needs.

IMR (Mitsubishi Heavy Industries, Japan)



Reactor type:	Integrated modular water reactor
Electrical capacity:	350 MW(e)
Thermal capacity:	1000 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Natural circulation
System pressure:	15.51 MPa
Core outlet temperature:	345°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	UO_2
Fuel enrichment:	4.8%
Fuel cycle:	26 months
Reactivity control:	Rod insertion
No. of safety trains:	4
Emergency safety systems:	Passive systems
Residual heat removal systems:	Active systems
Design life:	60 years
Design status:	Conceptual design completed
Seismic design:	Equivalent to that of PWRs in Japan
Predicted core damage frequency:	2.9E-7/reactor year
Planned deployment:	After 2020
Distinguishing features:	Steam generators in liquid and vapour regions of the vessel



IMR plant showing the stand-alone heat removal system and the hybrid heat transport system.

Introduction

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.

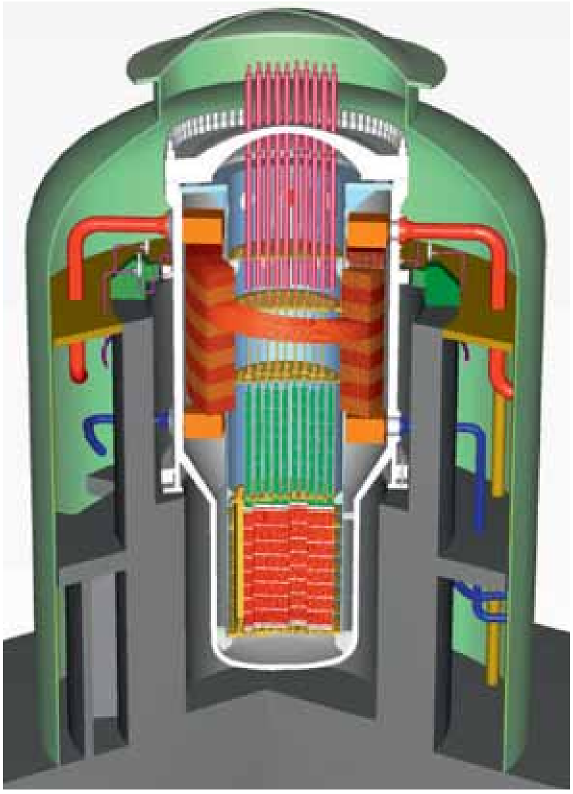
Description of the nuclear systems

The IMR has a core consisting of ninety-seven 21×21 fuel assemblies with an average enrichment of 4.95%. Control rods whose neutron absorber is 90wt% enriched B_4C perform the reactivity control, and a soluble acid boron system is used for the backup reactor shutdown to avoid corrosion of structural materials by boric acid. The hydrogen to uranium ratio (H:U) is set to five, which is larger than in conventional PWRs, to reduce the pressure drop in the primary circuit. The coolant boils in the upper part of the core and the core outlet void fraction is less than 20% locally and less than 40% in the core to keep bubbly flow conditions. To reduce axial power peaking caused by coolant boiling, the fuel consists of two parts: the upper part with higher enrichment and the lower part with lower enrichment. Additionally, hollow annular pellets are used in the upper part fuel to reduce axial differences of burnup rate.

The refuelling interval is 26 effective full-power months. The power density is about 40% of current PWRs but the fuel lifetime is 6.5 years longer, so that the average discharged burnup is about 46 GW·d/t, which is approximately the same as current PWRs. The cladding material employs Zr–Nb alloy to obtain integrity at a temperature of 345°C and over the long reactor lifetime.

Description of safety concept

The IMR is an LWR with moderation ratios similar to those of conventional LWRs, and, thus, its properties of fresh and spent fuel are also similar. This 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, pressurizers or coolant pipes with large diameters, nor does it have an emergency core cooling system 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, made possible by use of a stand-alone diesel generator. Due to the integrated primary system, the containment vessel is small, while simplified chemical and volume control and waste disposal systems are the result of boric acid-free operation.



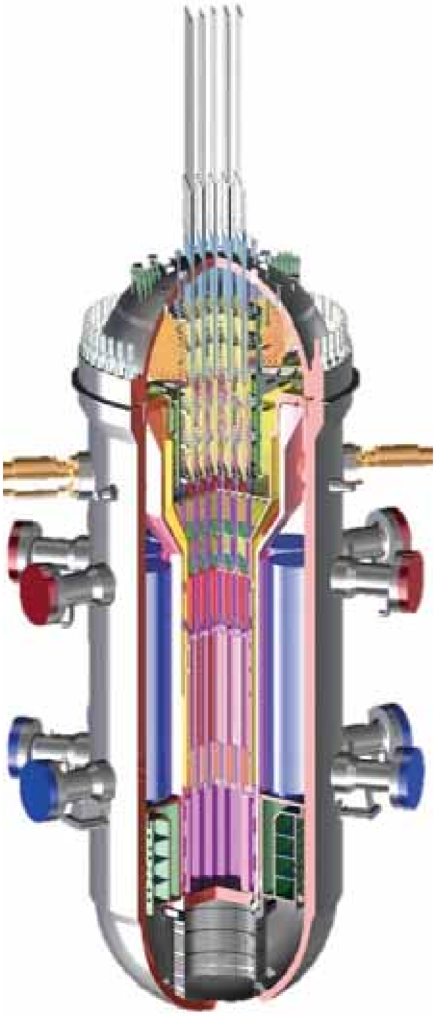
IMR containment layout.

Deployment status

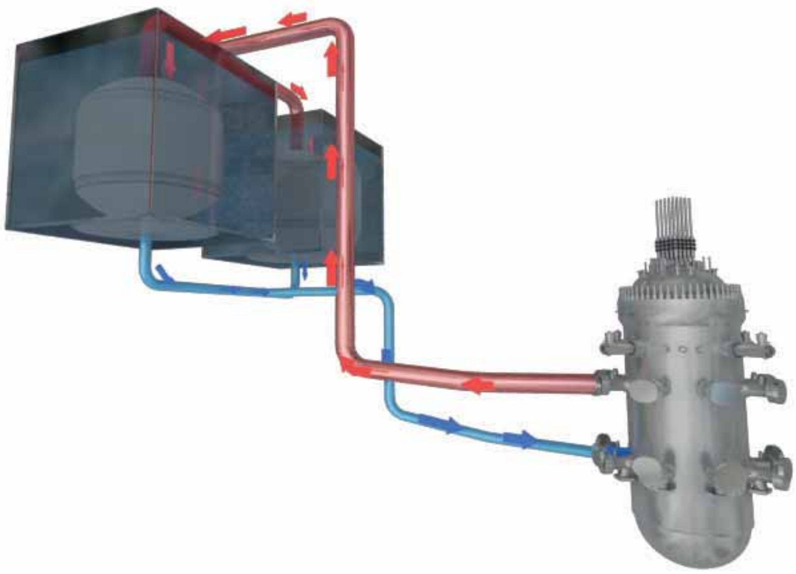
The IMR conceptual design study was initiated in 1999 by Mitsubishi Heavy Industries (MHI). A group led by MHI and including Kyoto University, the Central Research Institute of the Electric Power Industry and the Japan Atomic Power Company, developed related key technologies through two projects, funded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

Validation testing, research and development 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)



Reactor type:	Integral pressurized water reactor
Electrical capacity:	100 MW(e)
Thermal capacity:	330 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Forced circulation
System pressure:	15 MPa
Core outlet temperature:	323°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	UO_2
Fuel cycle:	36 months
Reactivity control:	Control rod drive mechanism, soluble boron and burnable poison
No. of safety trains:	4
Emergency safety systems:	Active and passive systems
Residual heat removal systems:	Passive systems
Design life:	60 years
Design status:	Standard design approval received in July 2012
Seismic design:	>0.18g automatic shutdown
Predicted core damage frequency:	1E-6/reactor year
Planned deployment:	2019
Distinguishing features:	Coupling of the desalination system or process heat application



Passive residual heat removal system.

Introduction

SMART (System Integrated Modular Advanced Reactor) is a small integral PWR with a rated thermal power of 330 MW(th). Its aims are 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.

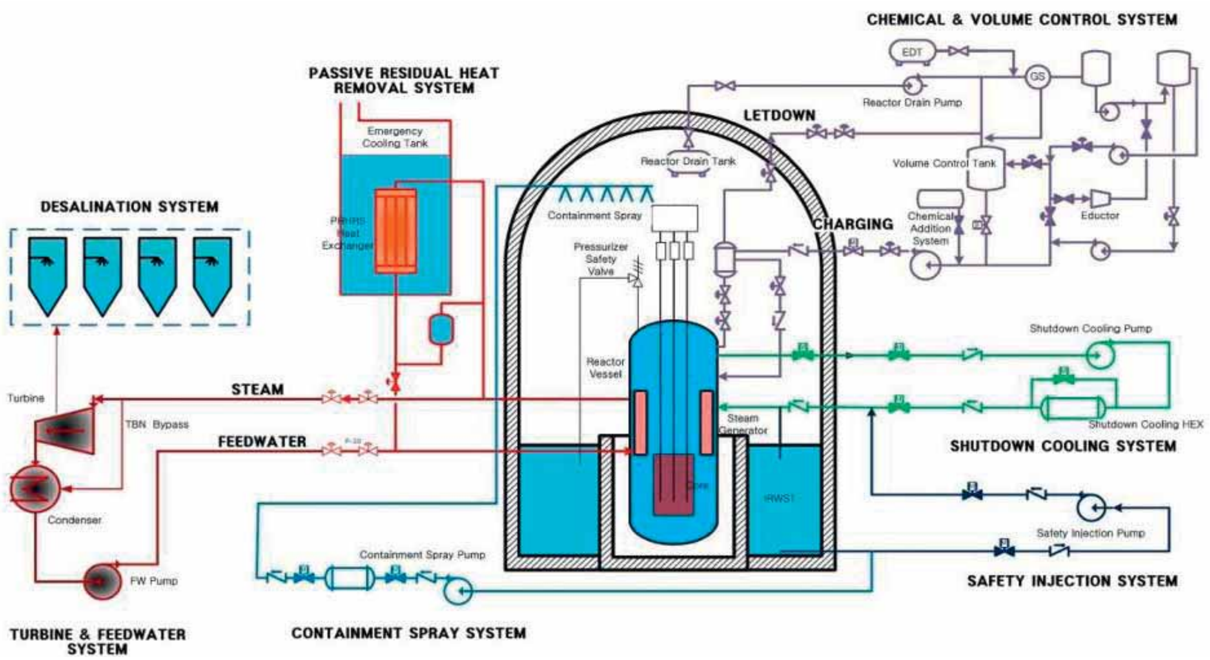
By introducing a passive RHRS removal system and an advanced loss of coolant accidents mitigation system, significant safety enhancement is expected by the design organization. The low power density design, with about a 5wt% UO_2 fuelled core, will provide a thermal margin of more than 15% to accommodate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

Description of the nuclear systems

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 36 months of 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.

Description of the safety concept

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 RHRS, 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.



Engineered safety systems.

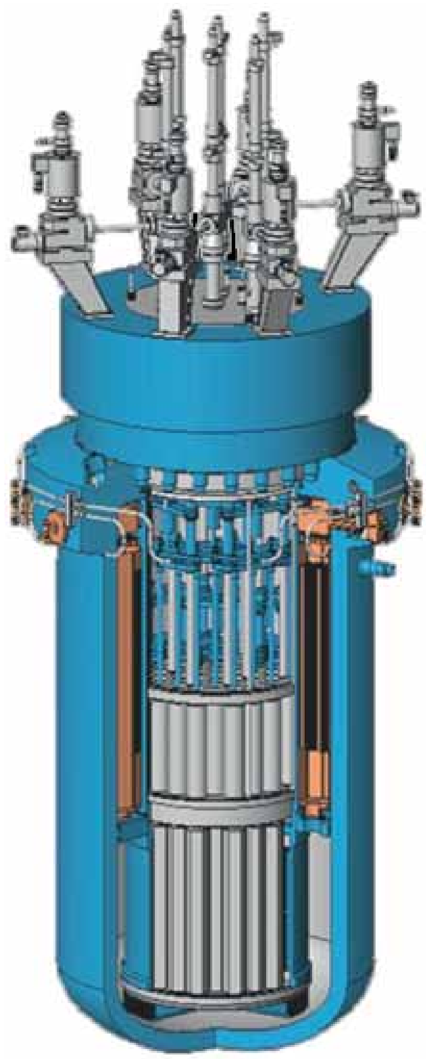
Description of the turbine-generator systems

The secondary system receives superheated steam from the nuclear steam supply system. It uses most of the steam for electricity generation and preheaters, and the remainder for non-electric applications. The sea water desalination system may be used in conjunction with the secondary system.

Deployment status

The standard design of SMART was approved by the Korean Nuclear Safety and Security Commission in July 2012. Construction is scheduled to begin before the end of 2012 and be completed by 2016.

ABV-6M (OKBM Afrikantov, Russian Federation)



Reactor type:	Pressurized water reactor
Electrical capacity:	8.6 MW(e)
Thermal capacity:	38 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Natural circulation
System pressure:	15.7 MPa
Core outlet temperature:	330°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	UO_2
Fuel enrichment:	19.7%
Core life:	10 years
Reactivity control:	Rod insertion
No. of safety trains:	N/A
Emergency safety systems:	N/A
Residual heat removal systems:	N/A
Design life:	60 years
Design status:	Detailed design
Seismic design:	N/A
Predicted core damage frequency:	N/A
Planned deployment:	N/A
Distinguishing features:	Natural circulation in the primary circuit for land based and floating nuclear power plants

Introduction

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 and recent achievements in the field of nuclear power plant 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.

Description of the nuclear systems

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 [4]. The reactor operates under the normal PWR conditions of 15.7 MPa in the reactor pressure vessel. The steam generators located inside the vessel generate 290°C steam at 3.14 MPa flowing at 55 t/h.

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

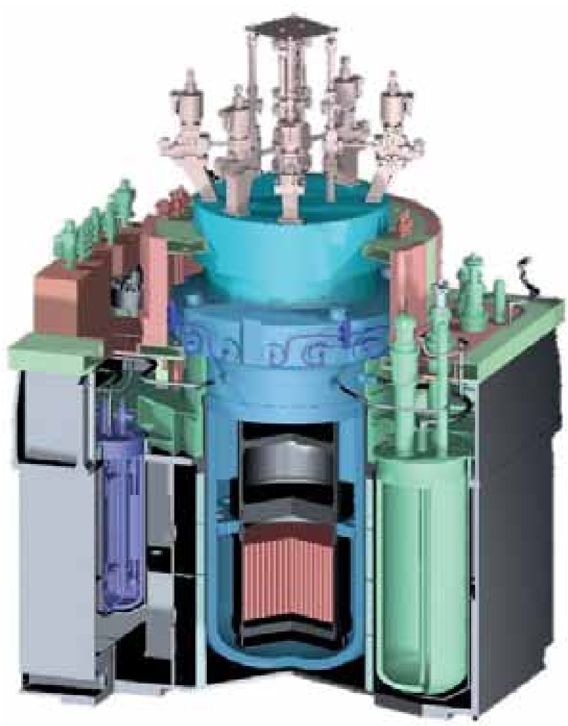
The main reactor pump equipment is arranged on the shield tank as a single steam generating aggregate (SGA) that is transportable by rail. The SGA has a mass of 200 t and is 5 m long by 3.6 m wide with a height of 4.5 m.

The core lifetime without reloading or shuffling of fuel is 10 years with 16 000 h of continuous operation.

Deployment capability

Specifically, the ABV reactor installation is intended as a universal power source for floating nuclear power plants. 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 [4]. Depending on the needs of the region, the floating nuclear power plant can generate electric power or provide heat and power cogeneration or heat generation for sea water desalination, or can be used for other applications.

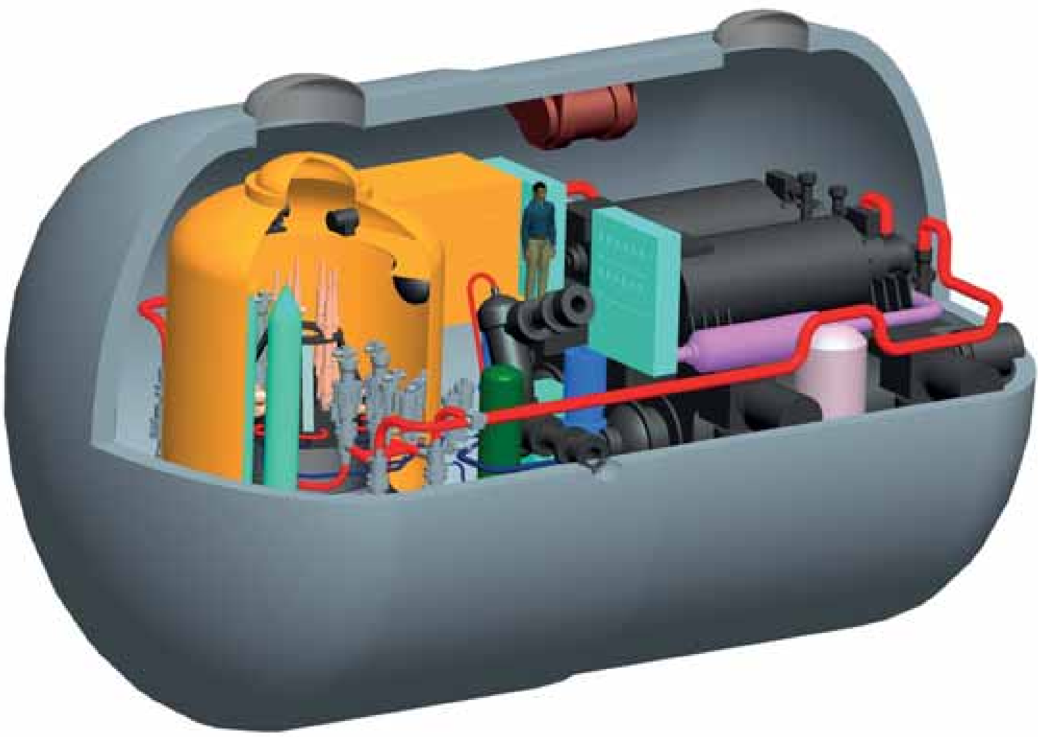
The stationary nuclear power plant — land based or underground — is fabricated as large, ready made units; these units are transported to the site in a special truck or by water. The total



ABV-6M reactor plant.

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 nuclear power plant is factory fabricated.

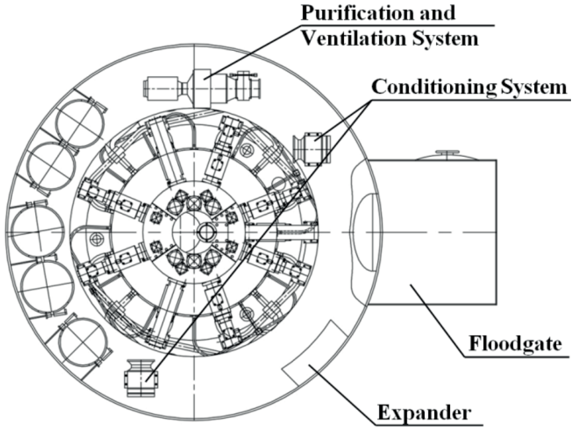
SHELF (NIKIET, Russian Federation)



Reactor type:	Pressurized water reactor
Electrical capacity:	6.0 MW(e)
Thermal capacity:	28 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Forced and natural circulation
System pressure:	17 MPa
Core outlet temperature:	320°C
Thermodynamic cycle:	Direct Rankine cycle
Fuel material:	UO_2 and aluminium alloy matrix
Fuel enrichment:	<20%
Fuel cycle:	56 months
Reactivity control:	Rod insertion
No. of safety trains:	2
Emergency safety systems:	Active
Residual heat removal systems:	Passive
Design life:	30 years
Design status:	Concept description
Seismic design:	7.0g
Predicted core damage frequency:	1E-6/reactor year
Planned deployment:	N/A
Distinguishing features:	Underwater energy source

Introduction

The N.A. Dollezhal Research and Development Institute of Power Engineering in the Russian Federation is currently developing a nuclear turbine-generator plant of 6 MW(e) as an underwater energy source. The plant comprises a two circuit nuclear reactor facility with a water cooled and water moderated reactor of 28 MW(th), a turbine-generator plant with a capacity of 6 MW, and an automated remote control, monitoring and protection system by means of engineered features, including electricity output regulation, control and monitoring instrumentation.



SHELF containment top view.

Description of the nuclear systems

The SHELF reactor uses an integral reactor with forced and natural circulation in the primary circuit, in which the core, steam generator (SG), motor-driven circulation pump and control and protection system drive are housed in a cylindrical vessel.

The core is of the heterogeneous pressure-tube type, with self-spacing fuel elements with a ^{235}U enrichment below 20%. The SGs are once-through helical coils, made of titanium alloys and separated into sections that have their own cover outlet for steam and feedwater and can be plugged when necessary.

The core, the compensation groups, the mounting plates and the radiation shields are installed in a removable screen. The first reactor core is loaded at the reactor manufacturer site by placement in a removable screen with continuous critical mass monitoring. After the loading is completed and the compensation rods are fixed in proper positions, the removable screen is installed into the reactor vessel.

Description of the safety concept

The reactor integral design has eliminated the primary coolant piping and the associated welded joints and valves, which, the designers claim, has given the installation a simpler design and increased reliability, strength and safety. The existing make-up pipelines and pressurizer connections are located inside the containment and the make-up line is shut off during operation. The cover has throttling devices (orifice plates) installed at the piping outlets for limiting leakage, if any. In the case of a leak, reactor safety is achieved by pressure levelling in the containment and in the reactor, and by termination of the outflow.

The components and pipelines, which are subjected to the primary circuit pressure and potential depressurization points, are kept inside the containment to confine any radioactive substances during accidents. The containment also includes the possibility of flooding the core to remove decay heat, along with a passive cooldown system for residual heat removal.

The SHELF reactor operates in an underwater, sealed capsule that is monitored and controlled from a coastal station or a floating structure, depending on the user's request.

Description of the turbine-generator systems

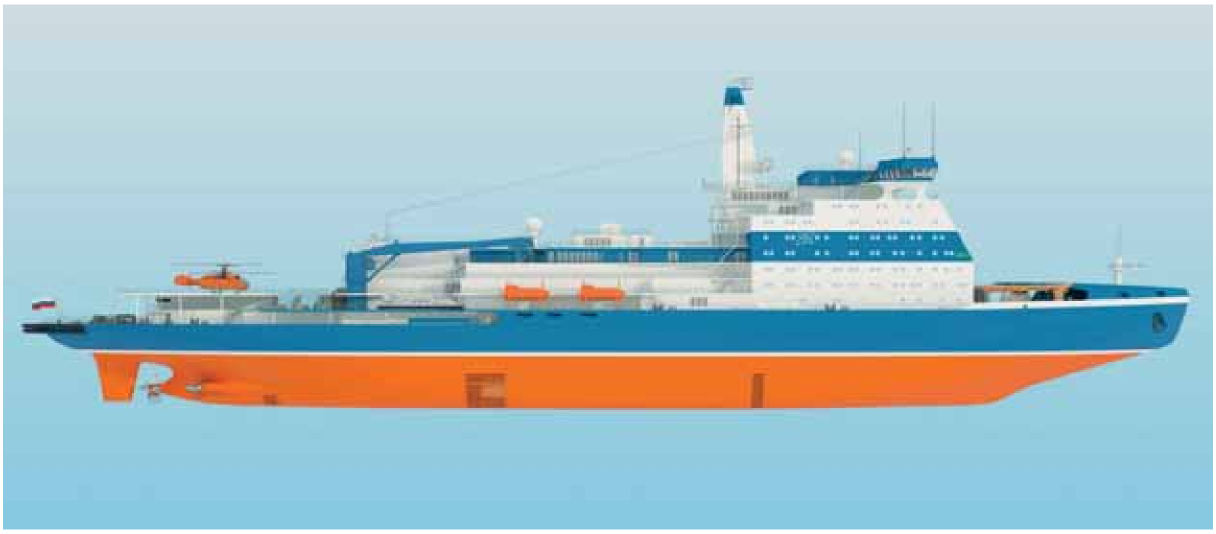
The turbine-generator designed for this plant will use water instead of oil in the lubrication and regulation systems as a solution to fire safety issues.

Deployment capability

At present, the SHELF reactor is in the early design phase and does not yet include a planned date of deployment.

RITM-200 (OKBM Afrikantov, Russian Federation)

Reactor type:	Integral pressurized water reactor
Electrical capacity:	50 MW(e)
Thermal capacity:	175 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Forced circulation
System pressure:	15.7 MPa
Core outlet temperature:	N/A
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	UO_2
Fuel enrichment:	<20%
Fuel cycle:	84 months
Reactivity control:	Rod insertion
No. of safety trains:	N/A
Emergency safety systems:	N/A
Residual heat removal systems:	N/A
Design life:	40 years
Design status:	Detailed design
Seismic design:	N/A
Predicted core damage frequency:	N/A
Planned deployment:	2017
Distinguishing features:	Developed for universal nuclear icebreakers



Universal atomic icebreaker project [5].

Introduction

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

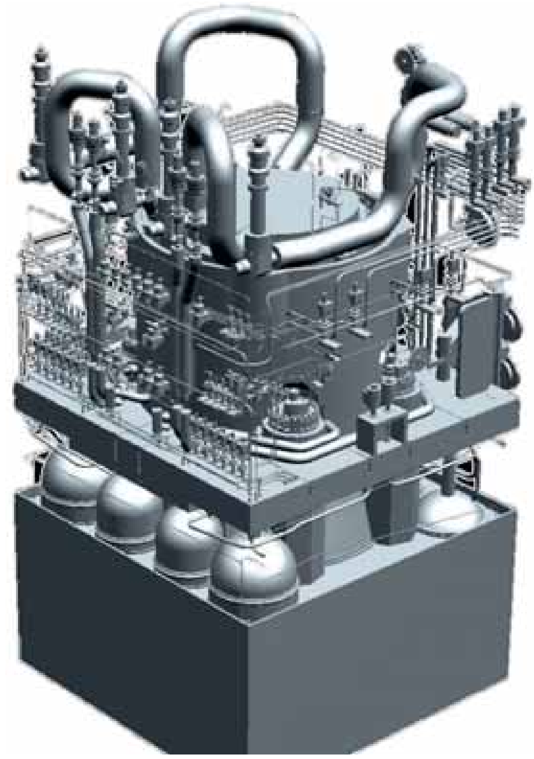
Description of the nuclear systems

It employs a low enriched cassette type reactor core similar in design to the KLT-40S. The fuel enrichment is up to 20wt% and the reactor houses 199 fuel assemblies. 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 steam generator (SG) cassette nozzles. The four section SGs have 12 rectangular cassettes, while the four main circulation pumps are located in the cold leg of the primary circulation path and 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/h.

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 transport 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.

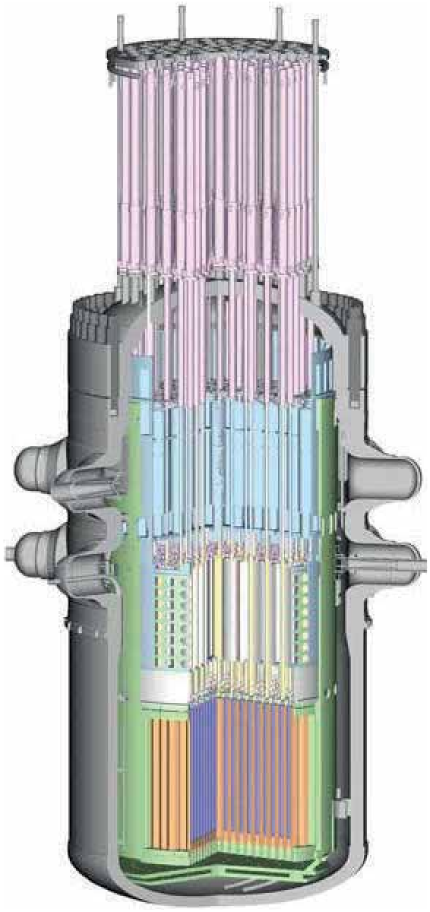


RITM reactor plant.

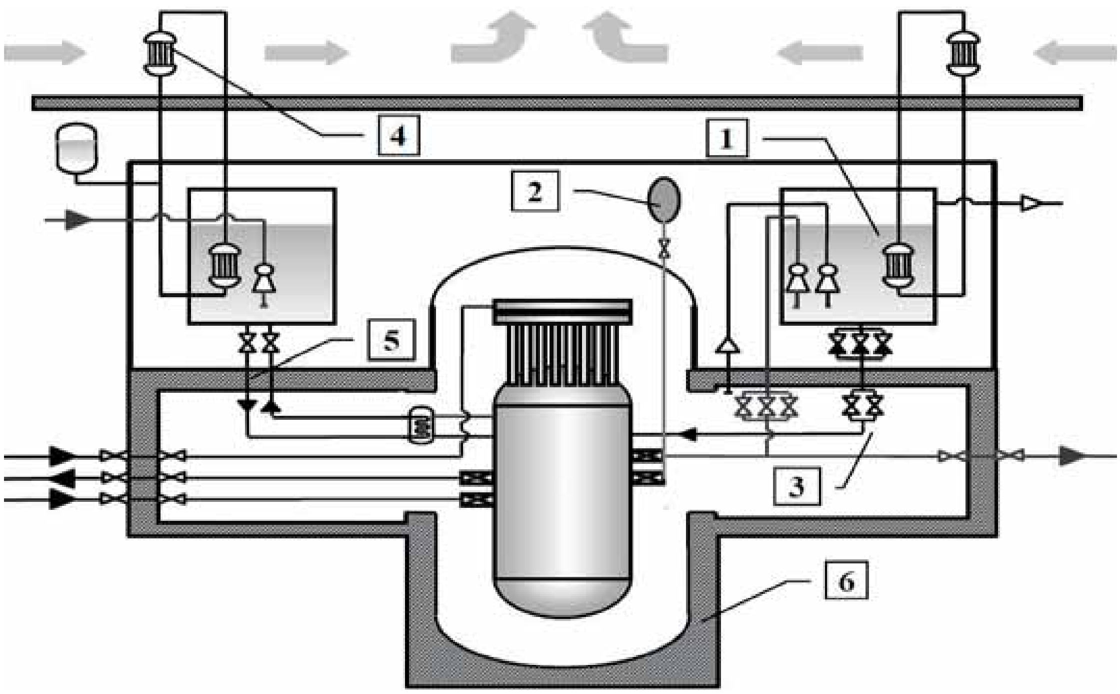
Deployment status

The designer is currently designing a versatile nuclear icebreaker called 'Project 22220' to be deployed using the RITM-200 nuclear plant [6]. No date has currently been set for deployment.

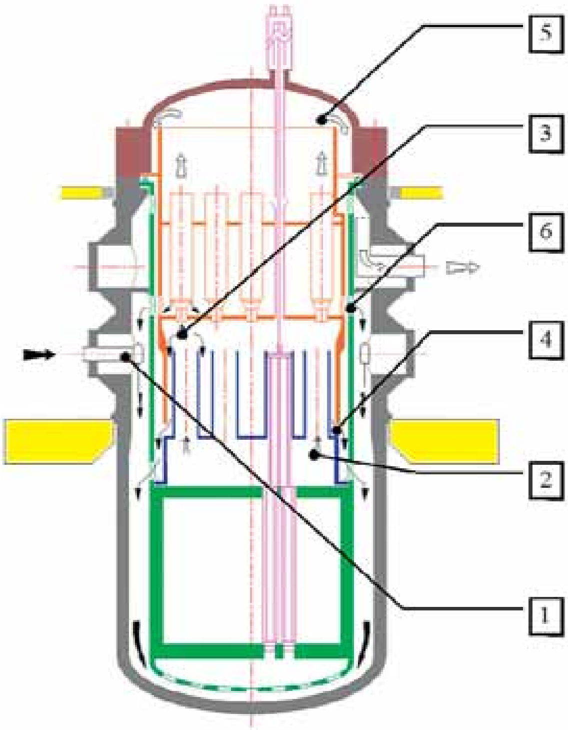
VK-300 (RDIPE, Russian Federation)



Reactor type:	Boiling water reactor
Electrical capacity:	250 MW(e)
Thermal capacity:	750 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Natural circulation
System pressure:	6.9 MPa
Core outlet temperature:	285°C
Thermodynamic cycle:	Direct Rankine cycle
Fuel material:	UO_2
Fuel enrichment:	4%
Fuel cycle:	18 months
Reactivity control:	Rod insertion
No. of safety trains:	2
Emergency safety systems:	Passive
Residual heat removal systems:	Passive
Design life:	60 years
Design status:	Conceptual design
Seismic design:	N/A
Predicted core damage frequency:	1E-7/reactor year
Planned deployment:	N/A
Distinguishing features:	Cogeneration BWR based on operating prototype



Reactor plant flow diagram showing the (1) emergency cooldown tank, (2) liquid absorber storage vessel, (3) emergency core flooding system, (4) air heat transfer system, (5) ECCS and (6) primary containment.



Reactor flow diagram showing (1) feedwater, (2) out-core mixing chamber, (3) preliminary separation chamber, (4) pre-separated water outlet, (5) steam and (6) major separated water stream.

Introduction

The VK-300 is a 250 MW(e) simplified water cooled and water moderated BWR with natural circulation of coolant and passive systems [2]. A VK-50 reactor operated in the Russian Federation for 31 years and has been used as a prototype for the VK-300 design [7].

Description of the nuclear systems

The VK-300 reactor has a small reactivity margin for nuclear fuel burnup thanks to partial overloading and use of burnable absorbers. It employs fuel elements as well as cyclone separators in the steam generators from the WWER-1000 class.

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.

Description of the safety concept

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 provide safety assurance economically and reliably using structurally simple, passive safety systems.

The emergency cooldown tanks (ECTs) are located outside of the PC and are intended to function as accumulators and primary inventory make-up. If there is a line rupture and the pressure of the PC and reactor equalize, the ECTs actuate by gravity and fill the PC.

The residual heat is passively removed from the reactor by steam condensers located in the PC around the reactor that are normally flooded with the primary circuit water. When the level in the PC drops, the connecting pipelines to the condensers are opened, the reactor steam condenses and it returns back to the reactor. The condensers are cooled with water from the ECTs.

Description of the turbine-generator systems

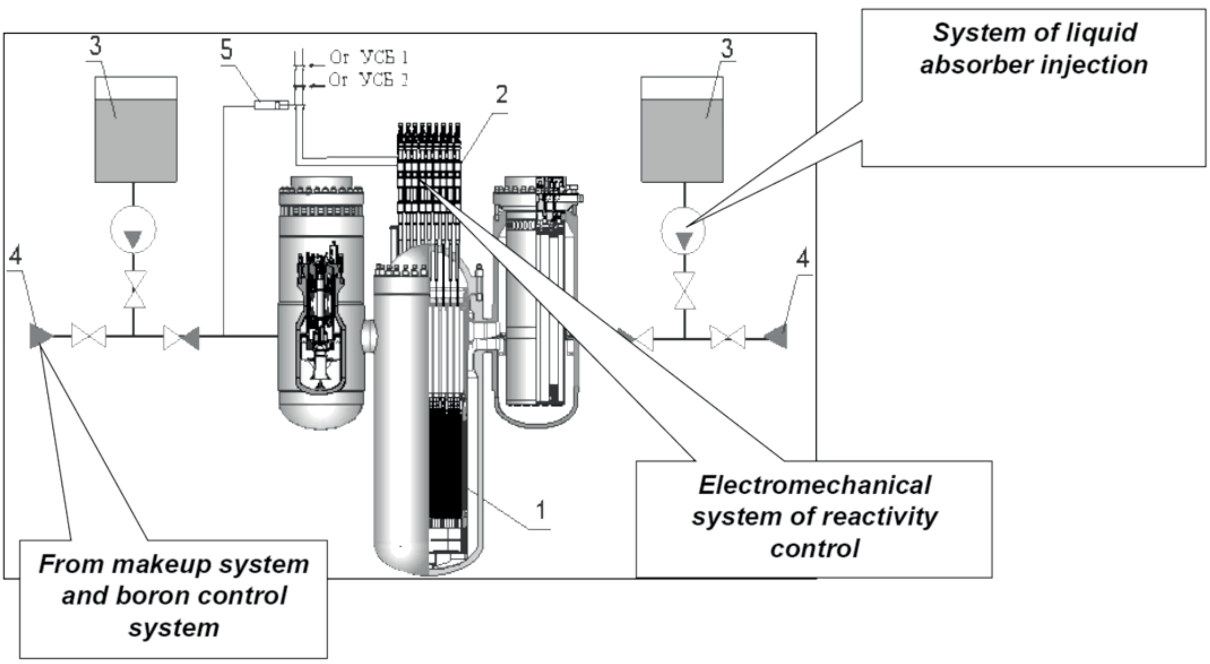
The design of the 750 MW(th), 250 MW(e) VK-300 was developed to supply electricity and heat of up to 465 MW(th) within a nuclear cogeneration plant to be built at the Krasnoyarsk Mining and Chemical Combine. This design was developed by Russian research and design organizations including the Research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

Deployment status

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
Electrical capacity:	325 MW(e)
Thermal capacity:	917 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Forced circulation
System pressure:	12.7 MPa
Core outlet temperature:	316°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	UO_2
Fuel enrichment:	4.95%
Fuel cycle:	72 months
Reactivity control:	Rod insertion and boron dilution
No. of safety trains:	2
Emergency safety systems:	Active and passive
Residual heat removal systems:	Active and passive
Design life:	60
Design status:	Conceptual design
Seismic design:	VIII MSK-64
Predicted core damage frequency:	1E-6/reactor year
Planned deployment:	2016
Distinguishing features:	Power source for floating nuclear power plants



Emergency shutdown systems.

Introduction

The VBER-300 reactor is a medium sized power source for land based nuclear power plants and nuclear cogeneration plants, as well as for floating nuclear power plants.

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 water cooled, water moderated power reactor (WWER) type reactors and achievements in the field of nuclear power plant safety. Features of the VBER-300 include the use of proven nuclear ship building technologies, engineering solutions and operating experience of WWER reactors as well as the possibility to enlarge or reduce the source power using only unified VBER-300 equipment (reactors consisting of two, three or five loops).

Description of the nuclear systems

The VBER-300 design concept allows a flexible fuel cycle for the reactor core with standard WWER fuel assemblies (FAs). The fuel cycles are 3 × 2 years and 4 × 1.5 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.

The main reactor coolant pump is an axial, single-stage canned motor pump with experience from operation of more than 1500 main circulation pumps on ships. The steam generators are once-through modules with the secondary coolant flowing within the tubes.

Description of the safety concept

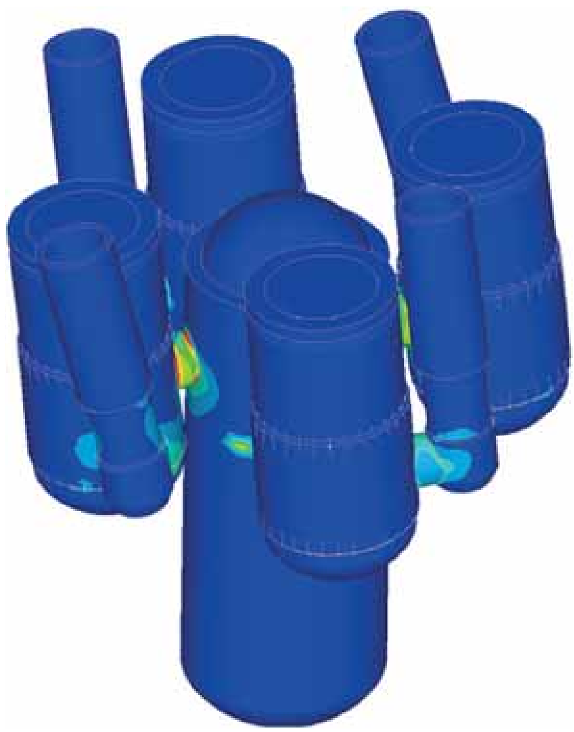
The safety assurance engineering solutions incorporated into the design are: prioritization of accident prevention measures, design simplification, inherent safety, and the defence in depth principle; passive safety systems and enhancement of safety against external impacts (including acts of terrorism); and limitation of severe accident consequences.

The VBER-300 emergency shutdown system consists of the control rod drives, two trains of liquid absorber injection, and two trains of boron control from the make-up system. The emergency core cooling system consists of two stages of hydraulic accumulators and the RHRS consists of two passive emergency heat removal systems and a process condenser.

The estimated reactor unit strength holds up against seismic impacts of maximum magnitude VIII on the MSK-64 scale.

Deployment status

The VBER-300 preliminary design was completed in 2002, and a technical and commercial proposal



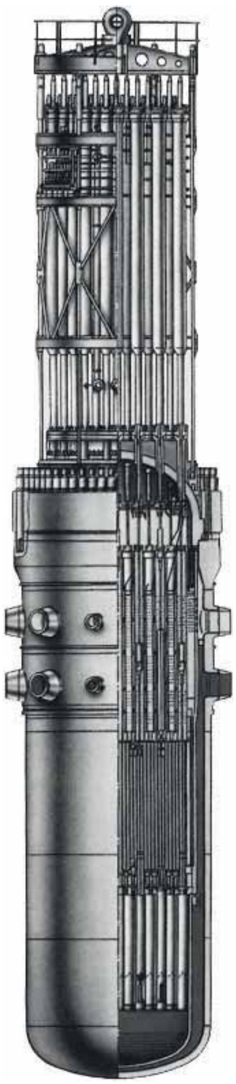
Seismic stability simulation.

(a shorter version of technical and economic investigation) for construction of a land based or floating nuclear power plant with the VBER-300 was prepared.

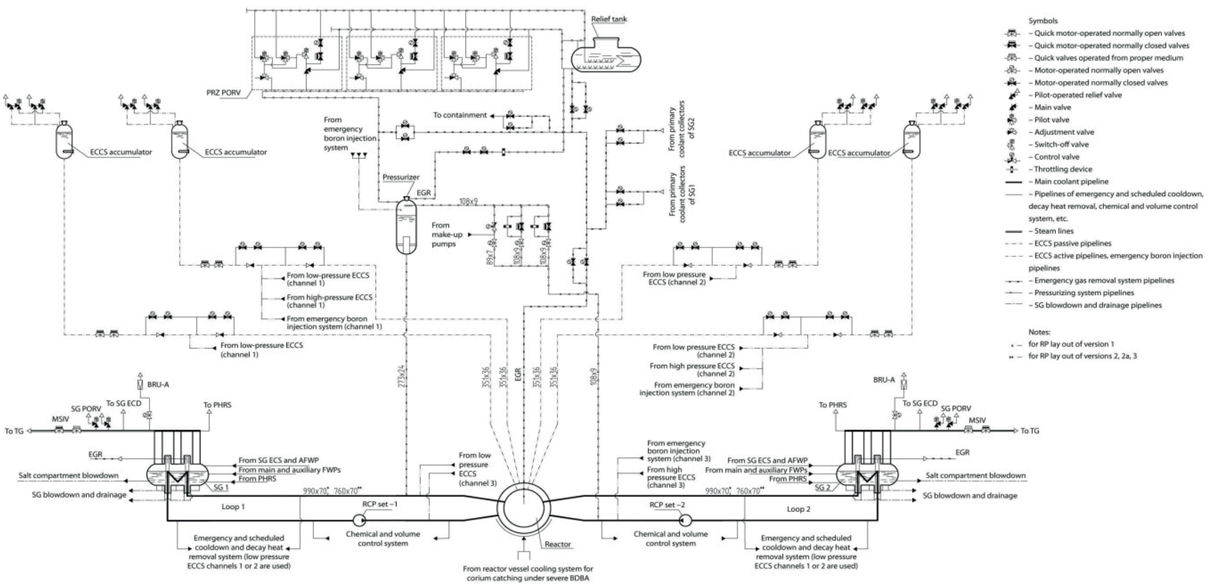
The preliminary design passed the branch review by Rosatom and was approved by the Scientific and Technical Council. Currently, there are two directions of further project development: first, within the framework of scientific and technical cooperation between the Russian Federation and Kazakhstan, and second, replacement of outdated nuclear power plant capabilities or construction of new medium sized nuclear power plants in the Russian Federation.

Kazatomprom JSC of Kazakhstan completed a feasibility study for the VBER-300 in Aktau city in 2009 and plans to complete construction of the first unit by 2016 [8].

WWER-300 (OKBM Gidropress, Russian Federation)



Reactor type:	Pressurized water reactor
Electrical capacity:	300 MW(e)
Thermal capacity:	850 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Forced circulation
System pressure:	16.2 MPa
Core outlet temperature:	325°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	UO_2
Fuel enrichment:	4.95%
Fuel cycle:	72 months
Reactivity control:	Rod insertion and soluble boron
No. of safety trains:	4
Emergency safety systems:	Active and passive
Residual heat removal systems:	Active and passive
Design life:	60 years
Design status:	Detailed design
Seismic design:	VII–MSK 64
Predicted core damage frequency:	1E-6/reactor year
Planned deployment:	N/A
Distinguishing features:	Based on the operating experience of various WWER type reactor designs



Reactor core flow diagram.

Introduction

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 fuel assemblies (FAs) similar to those used in the WWER-1000.

Description of the nuclear systems

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 steam generator (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.

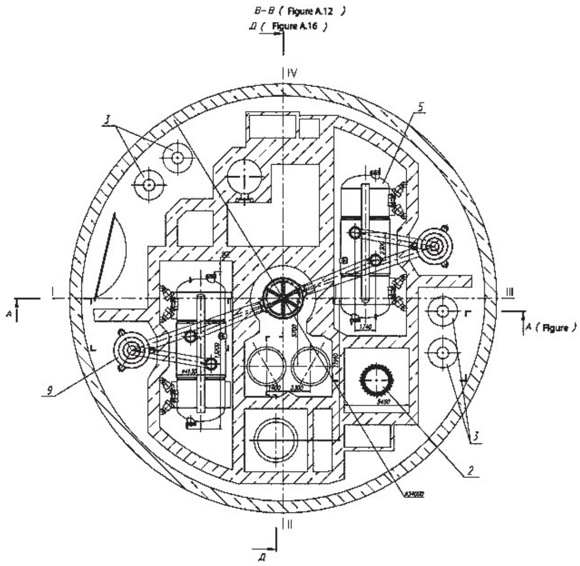
The feedwater system comprises the main feedwater pumps, standby feedwater pump, de-aerator, isolation and control valves, and pipelines. The feedwater supply is provided by three feedwater pumps from the de-aerator plant.

Description of the safety concept

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 nuclear power plants. The safety features used in the V-478 design include: SG emergency cooldown systems, emergency gas removal systems, emergency boron injection system, reactor primary coolant and spent fuel pool emergency and scheduled cooldown system, a main steam line isolation system, emergency passive core cooling system, high and low pressure core cooling pumps, double containment, core catcher and passive heat removal.

The passive emergency core cooling system (ECCS) injects the boric acid solution with a concentration not below 16 g/kg into the reactor at a primary pressure below 5.9 MPa. The supply should be sufficient for core heat removal before connection of the low pressure part of the reactor primary coolant and spent fuel pool emergency and scheduled cooldown systems under a design basis loss of coolant accidents (LOCAs).

The high pressure ECCS is proposed for core heat removal under a primary LOCA when the compensation capacity of the normal make-up water is not sufficient or is unavailable. The low pressure ECCS is proposed for decay heat removal.



Reactor plant layout showing the (1) reactor coolant pump set, (2) pressurizer, (3) emergency core cooling system accumulators, (4) main coolant pipeline, (5) steam generator and (6) the reactor.

Deployment status

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 nuclear power plant are proved by the more than 1400 reactor years of operating experience of WWER plants.

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

Research and development 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
Electrical capacity:	35 MW(e)
Thermal capacity:	150 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Forced circulation
System pressure:	12.7 MPa
Core outlet temperature:	316°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	UO_2
Fuel enrichment:	<20%
Fuel cycle:	28 months
Reactivity control:	Rod insertion
No. of safety trains:	2
Emergency safety systems:	Active and passive systems
Residual heat removal systems:	Passive system
Design life:	40 years
Design status:	Under construction
Seismic design:	3.0g
Predicted core damage frequency:	1E-6/reactor year
Planned deployment:	2012
Distinguishing features:	Floating power unit for heat and electricity

Possible floating power unit configuration.

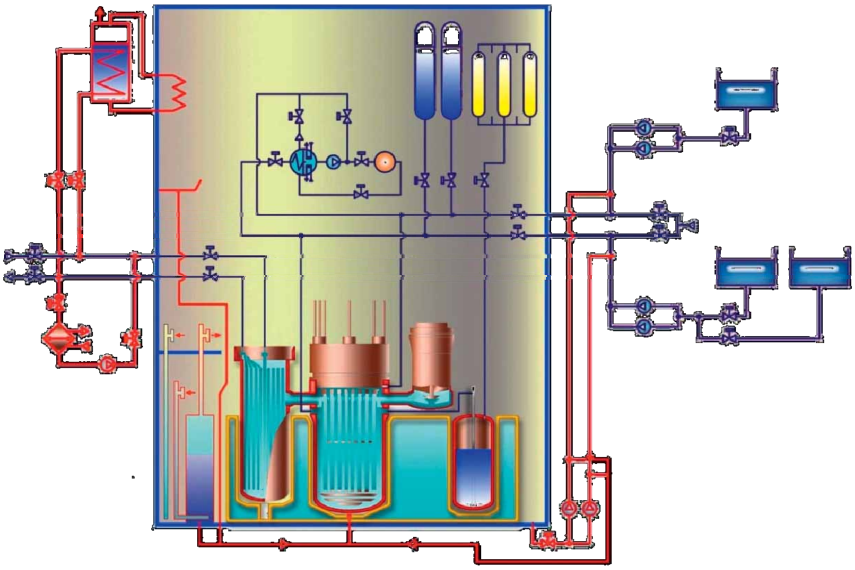
Introduction

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

The floating nuclear power plant with a KLT-40S reactor can be manufactured in shipyards and can then be 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 nuclear power plants, and there is a high degree of freedom in selecting the location for a floating nuclear power plant as it can be moored in any coastal region. It also has 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 nuclear power plant maintenance and refuelling.

Fuel utilization efficiency is provided by the use of all of the 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.

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 is refuelled. The spent nuclear fuel is then stored on board the FPU, and no special maintenance or refuelling ships are necessary.



KLT-40S flow diagram.

Description of the nuclear systems

The reactor has a modular design with the core, steam generators (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.

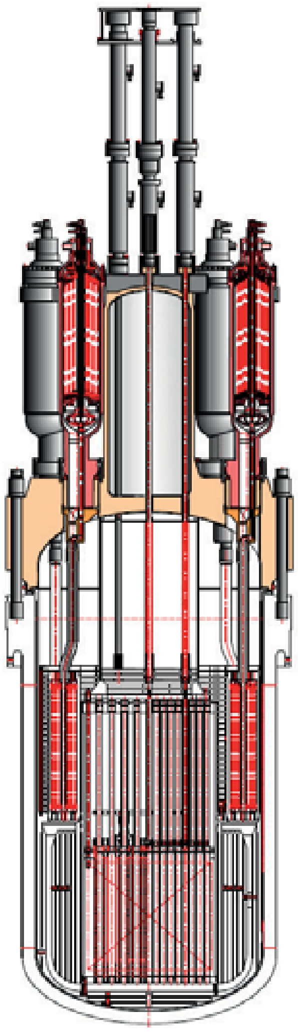
Description of the safety concept

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.

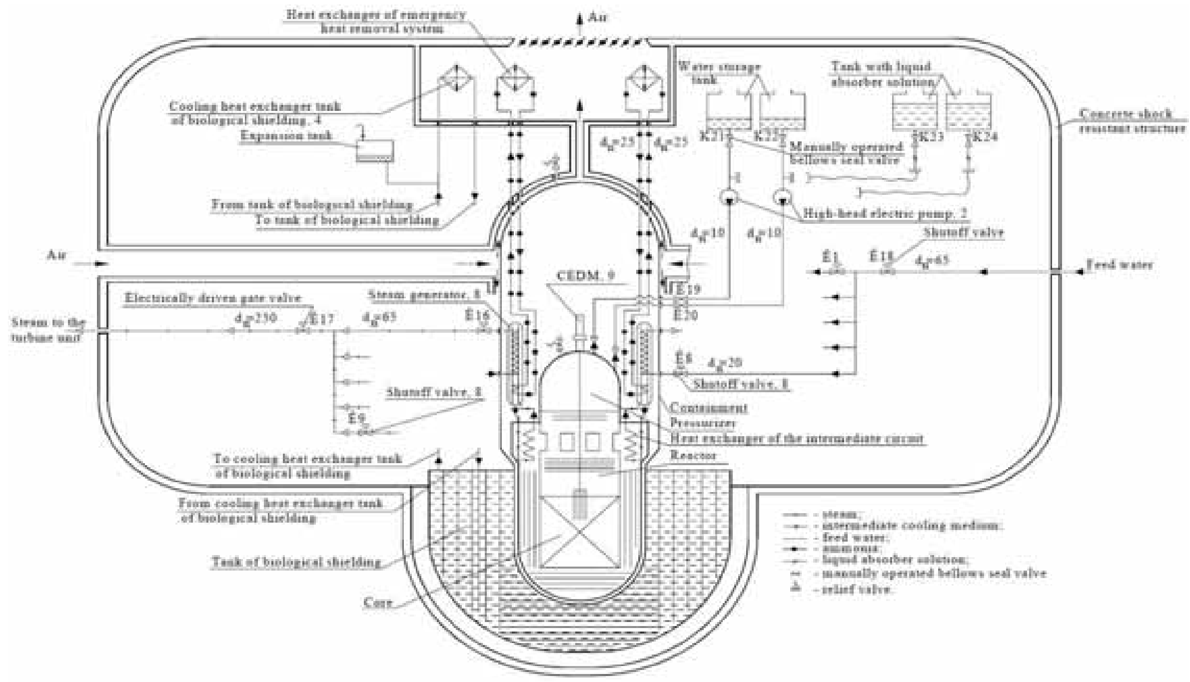
Deployment status

The keel of the first FPU carrying the KLT-40S, the Akademik Lomonosov, was laid in 2007. The Akademik Lomonosov is expected to be complete by the end of 2012 [9].

UNITHERM (RDIPE, Russian Federation)



Reactor type:	Pressurized water reactor
Electrical capacity:	2.5 MW(e)
Thermal capacity:	20 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Natural circulation
System pressure:	16.5 MPa
Core outlet temperature:	330°C
Thermodynamic cycle:	Direct Rankine cycle
Fuel material:	UO_2–ZrO_2 (CERMET)
Fuel enrichment:	19.75%
Fuel cycle:	25 years
Reactivity control:	Soluble boron and rod insertion
No. of safety trains:	2
Emergency safety systems:	Passive
Residual heat removal systems:	Passive
Design life:	25 years
Design status:	Conceptual design
Seismic design:	VIII–IX–MSK 64
Predicted core damage frequency:	N/A
Planned deployment:	N/A
Distinguishing features:	Unmanned reactor operation



UNITHERM schematic diagram.

Introduction

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

Description of the nuclear systems

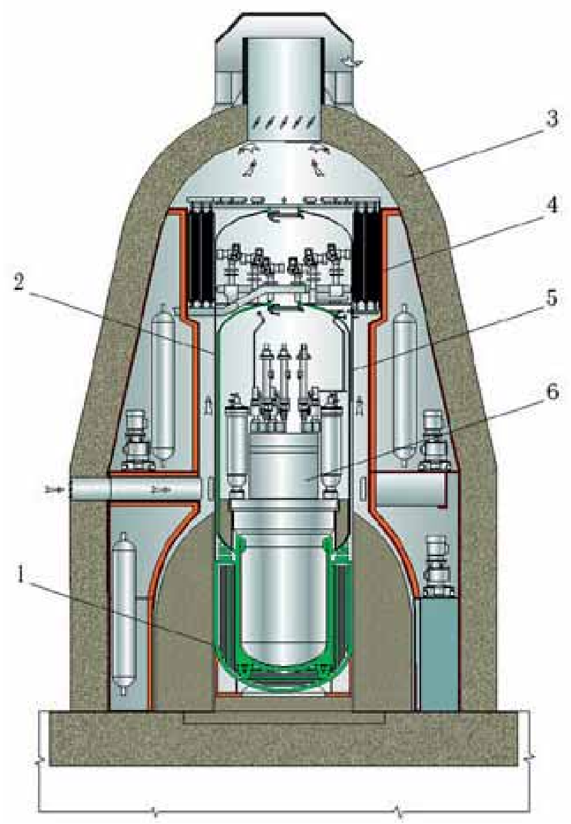
The UNITHERM fuel element is designed as a cylindrical rod with four spacing ribs on its outer surface. The fuel is in the form of tiny blocks of UO_2 grains coated with zirconium and dispersed in a zirconium matrix. The gap between the fuel-containing matrix and the cladding is filled with silumin. 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 fuel assemblies 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. Optimally shaped cladding is formed when the cladding is filled with the matrix composition.

Description of the safety concept

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 UNITHERM reactor facility showing the (1) iron-water shielding tank, (2) containment, (3) shock-proof casing, (4) cooldown system heat exchanger, (5) safeguard vessel and (6) the reactor.

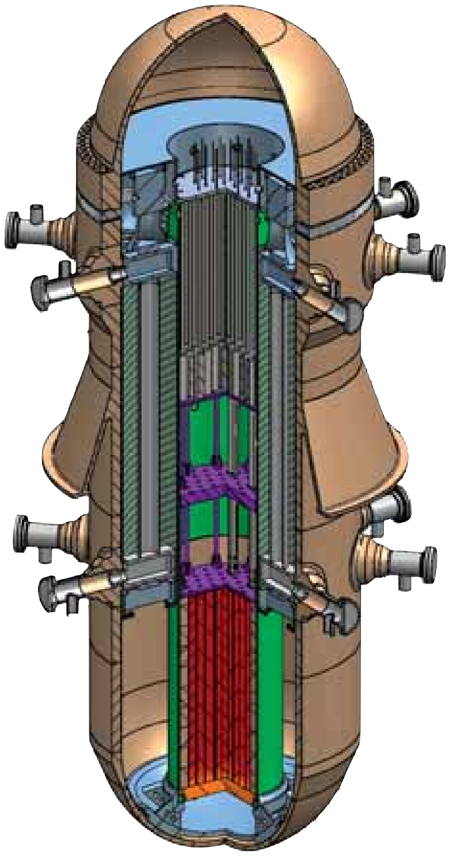
Description of the turbine-generator systems

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

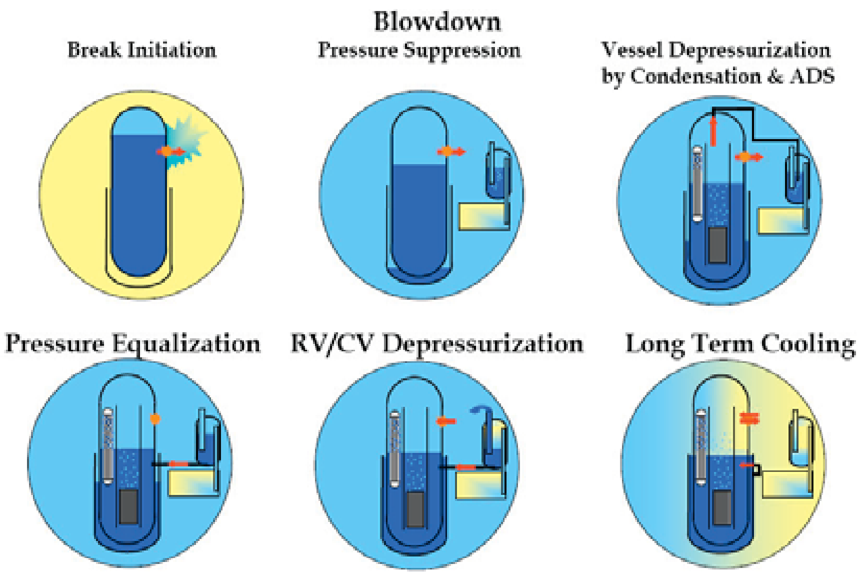
Deployment status

The UNITHERM nuclear power plant requires no major research and development for technology development. The detailed design stage would include qualification of the core, heat exchangers, CEDMs and other components [10].

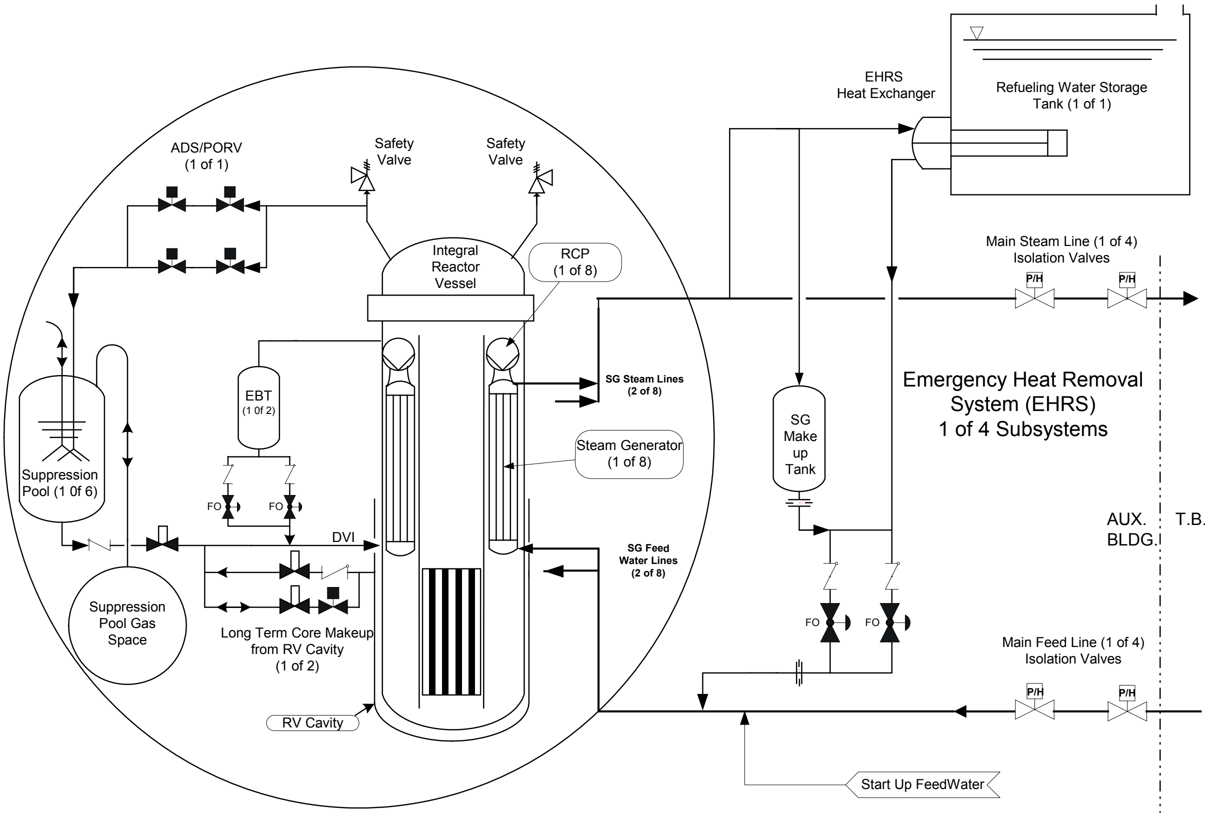
IRIS (IRIS, International Consortium)



Reactor type:	Integral pressurized water reactor
Electrical capacity:	335 MW(e)
Thermal capacity:	1000 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Forced circulation
System pressure:	15.5 MPa
Core outlet temperature:	330°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	UO_2/MOX
Fuel enrichment:	4.95%
Fuel cycle:	48 months
Reactivity control:	Soluble boron and rod insertion
No. of safety trains:	4
Emergency safety systems:	Passive
Residual heat removal systems:	Passive
Design life:	60 years
Design status:	Basic design
Seismic design:	0.3g
Predicted core damage frequency:	1E-8/reactor year
Planned deployment:	Currently seeking partnership for further development
Distinguishing features:	Integral primary system configuration



SBLOCA safety strategy.



Engineered safety features.

Introduction

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 steam generators, pressurizer and pumps;
- An effective safety approach;
- Optimized maintenance with intervals of at least four years.

Description of the nuclear systems

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. Fuel assemblies 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.

Description of the safety concept

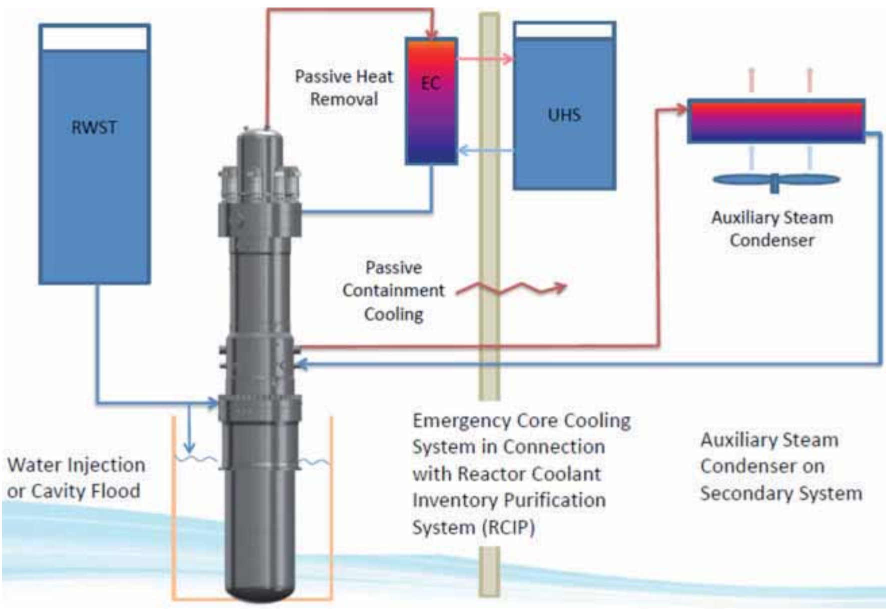
IRIS adopts passive safety systems and the safety by design philosophy including the risk informed approach. Due to IRIS's integral configuration by design (i.e. with no intervention of either active or passive systems), a variety of accidents either are eliminated or their consequences and/or probability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

Deployment status

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

mPower (Babcock & Wilcox, USA)

Reactor type:	Integral pressurized water reactor
Electrical capacity:	180 MW(e)
Thermal capacity:	530 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Forced circulation
System pressure:	14.1 MPa
Core outlet temperature:	320°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	UO_2
Fuel enrichment:	<5.0%
Fuel cycle:	48 months
Reactivity control:	Rod insertion
No. of safety trains:	N/A
Emergency safety systems:	Passive
Residual heat removal systems:	Passive
Design life:	60 years
Design status:	Basic design
Seismic design:	N/A
Predicted core damage frequency:	N/A
Planned deployment:	2020
Distinguishing features:	Internal once-through steam generator, pressurizer and control rod drive mechanism



Decay heat removal strategy.

Introduction

The Babcock & Wilcox (B&W) mPower™ reactor module is an integral PWR designed by B&W to generate an output of 180 MW(e).

Description of the nuclear systems

The reactor core consists of 69 fuel assemblies (FAs) that have less than 5% enrichment, Gd_2O_3 spiked rods, Ag In–Cd (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 to an active length of 241.3 cm and optimized specifically for the mPower reactor.

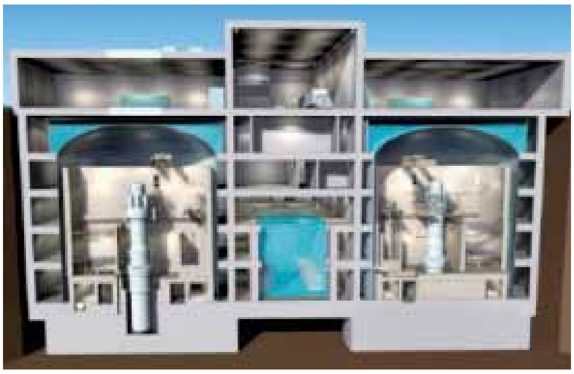
The reactor uses eight internal coolant pumps with external motors driving 3.8 m^3/s of primary coolant through the core. The integrated pressurizer at the top of the reactor is electrically heated and the reactor coolant pressure is nominally 14.1 MPa.

Description of the safety concept

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 loss of coolant accident (LOCA). The emergency core cooling system is connected with the reactor coolant inventory purification system and 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 uncovering 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.

The mPower reactor deploys a decay heat removal strategy with a passive heat exchanger connected with the ultimate heat sink, an auxiliary steam condenser on the secondary system, water injection or cavity flooding using the reactor water storage tank, and passive containment cooling.



mPower containment design.

Electrical, and instrumentation and control systems

The mPower reactor is being designed with digital instrumentation and control systems. The system is being designed with a high level of plant automation, including control of startup, shutdown and load following. The digital control system architecture is currently being developed.

Description of the turbine-generator systems

The balance of plant design consists of a conventional power train using a steam cycle and an optional air or water cooled condenser. The condenser determines the final power output of the plant; the water cooled condenser allows for an output power of 180 MW(e), while deploying an air cooled condenser would allow for an output of 155 MW(e).

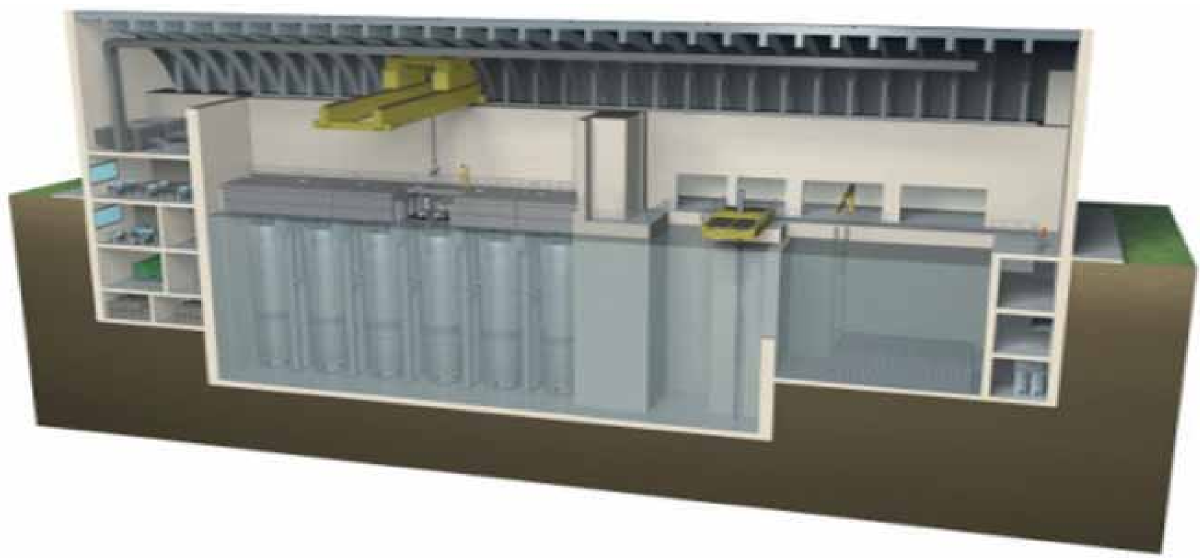
Deployment status and planned schedule

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 [11].

B&W submitted an application to the US Department of Energy for the SMR development support programme in March 2012 and is awaiting the results, expected by the end of 2012 [1].

NuScale (NuScale Power Inc., USA)

Reactor type:	Integral pressurized water reactor
Electrical capacity:	45 MW(e)
Thermal capacity:	160 MW(t)
Coolant/moderator:	Light water
Primary circulation:	Natural circulation
System pressure:	8.72 MPa
Core outlet temperature:	329°C
Thermodynamic cycle:	Indirect Rankine
Fuel material:	UO_2
Fuel enrichment:	<4.95%
Fuel cycle:	24 months
Reactivity control:	Rod insertion
No. of safety trains:	Two trains
Emergency safety systems:	Passive
Residual heat removal systems:	Passive
Design life:	60 years
Design status:	Basic design
Seismic design:	0.5g
Predicted core damage frequency:	1E-8/reactor year
Planned deployment:	2020
Distinguishing features:	Synergy through plant simplicity; reliance on existing light water technology and availability of an integral test facility



NuScale plant layout.

Introduction

In 2007, NuScale Power Inc. was formed to commercialize the concept of a plant that 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.

Description of the nuclear systems

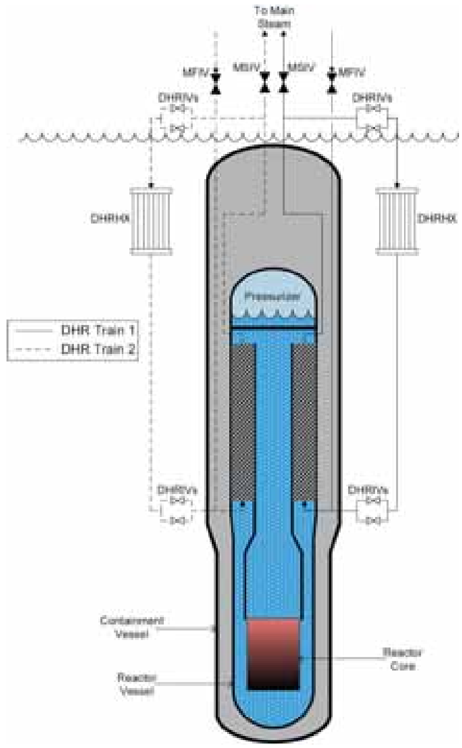
The designer claims that the NuScale reactor core contains 5% of the amount of fuel in large reactors. UO_2 fuel is used at an enrichment of less than 4.95% in a 17 × 17 fuel assembly with an active height of 2.0 m. The reactor has a fuel cycle lasting 24 months.

Description of the safety concept

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 NuScale reactor module operates solely on natural convection and resides in a high strength stainless steel containment vessel. The decay heat removal system shown in the following figure consists of two independent trains operating under two-phase natural circulation in a closed loop. The designers claim that the pool surrounding the reactor module provides three days of cooling supply for decay heat removal. The stainless steel containment also provides a decay heat removal capability by first venting reactor vessel steam, steam condensing on the containment, the condensate collecting in the lower containment region, and reactor recirculation valves opening to provide recirculation through the core. This is said to provide 30 days or more of cooling followed by indefinite air cooling.

The multi-module NuScale plant spent fuel pool 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.



Decay heat removal system.

Electrical, and instrumentation and control systems

The current NuScale design proposes using digital controls for the main control room and one operator controlling four reactor modules. Comprehensive human factor engineering and human–system interface studies are underway to determine the optimum number of reactors that can be effectively and safely controlled by a single operator.

Description of the turbine-generator systems

There are individual turbines for each of the reactor modules that are skid mounted and standard models currently available.

Deployment status and planned schedule

The NuScale Integral System test facility is being used to evaluate design improvements, and to conduct integral system tests for NRC certification.

NuScale Power Inc. submitted an application to the US Department of Energy for the SMR development support programme in March 2012 and is awaiting the results, expected by the end of 2012 [1].

Westinghouse SMR (Westinghouse, USA)

Reactor type:	Integral pressurized water reactor
Electrical capacity:	225 MW(e)
Thermal capacity:	800 MW(th)
Coolant/moderator:	Light water
Primary circulation:	Forced circulation
System pressure:	15.5 MPa
Core outlet temperature:	310°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	UO_2
Fuel enrichment:	<5.0%
Fuel cycle:	24 months
Reactivity control:	Soluble boron and rod insertion
No. of safety trains:	N/A
Emergency safety systems:	Passive
Residual heat removal systems:	Passive
Design life:	N/A
Design status:	Basic design
Seismic design:	N/A
Predicted core damage frequency:	N/A
Planned deployment:	N/A
Distinguishing features:	Incorporates passive safety systems and proven components of the AP1000

Internal control rod drive mechanism testing.

Introduction

Westinghouse officially introduced the Westinghouse SMR design in February 2011, and is currently preparing for a role in the US Department of Energy's demonstration programme. The Westinghouse SMR is a 200 MW(e) class integral PWR with all primary components located inside the reactor vessel.

Description of the nuclear systems

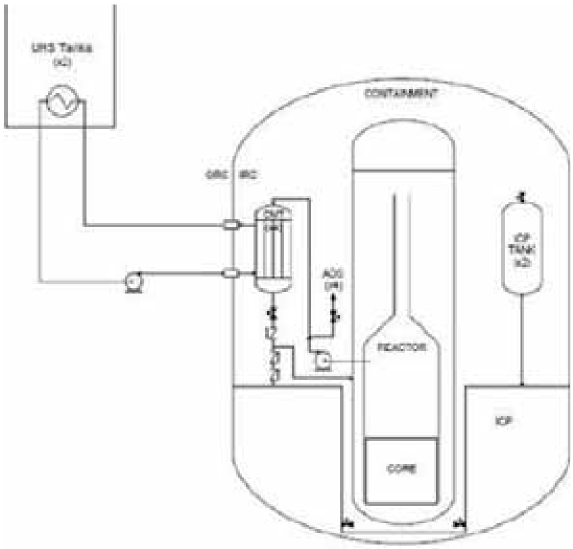
The reactor core is a partial height version of the 17 × 17 fuel assembly design used in the AP1000 reactor. The active core height is 2.4 m with 89 fuel assemblies and 37 internal control rod drive mechanisms (CRDMs). The fuel is 5% enriched and the core is designed to be refuelled every 24 months.

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 steam generators are recirculating, once-through straight tubes, which achieves 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, and 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.



Decay heat removal strategy.

Description of the safety concept

The Westinghouse SMR utilizes four heat exchangers connected to two ultimate heat sinks for decay heat removal.

Electrical, and instrumentation and control systems

The instrumentation and control system for the Westinghouse SMR is the OVATION-based digital control system.

Description of the turbine-generator systems

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.

Deployment status

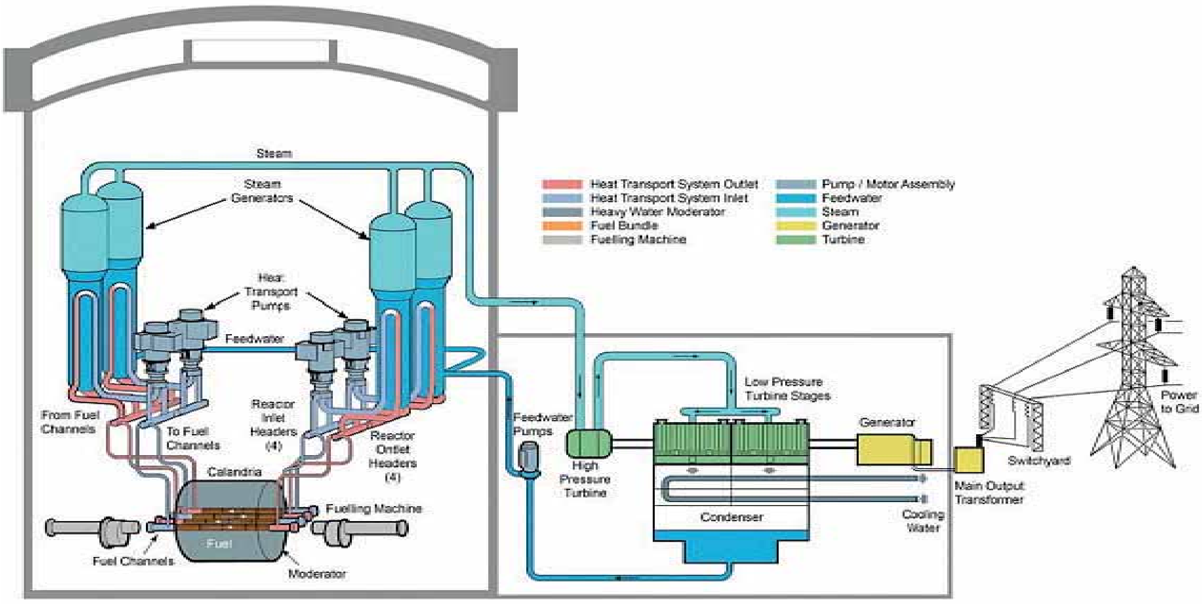
Westinghouse submitted an application to the US Department of Energy for the SMR development support programme in March 2012 and is awaiting the results, expected by the end of 2012 [1].

HEAVY

WATER

REACTORS

EC6 (AECL, Canada)



Reactor type:	Pressure tube type reactor
Electrical capacity:	740 MW(e)
Thermal capacity:	2084 MW(th)
Coolant/moderator:	Heavy water (D_2O)
Primary circulation:	Forced circulation
System pressure:	10.09 MPa
Core outlet temperature:	310°C
Thermodynamic cycle:	Indirect Rankine
Fuel material:	UO_2
Fuel enrichment:	Natural uranium
Fuel cycle:	Closed cycle
Reactivity control:	Mechanical control absorbers, adjusters, soluble poison
No. of safety trains:	2
Emergency safety systems:	Active and passive systems
Residual heat removal systems:	Active and passive systems
Design life:	60 years
Design status:	Basic design
Seismic design:	0.3g
Predicted core damage frequency:	1E-6/reactor year
Planned deployment:	N/A
Distinguishing features:	Based on proven CANDU experience

Introduction

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.

Description of the nuclear systems

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.

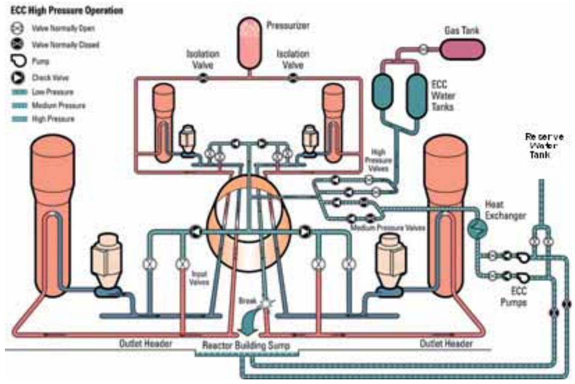
The fuel bundle consists of 37 elements. Each element contains three basic components: the UO_2 pellets, the sheath with canlub (graphite) coating in the inside surface and the end caps. The designers claim that various fuel cycle options can be accommodated in the EC6, including slightly enriched uranium up to 1.2%, thorium, mixed oxide fuel and natural uranium equivalent fuel, which can be produced from recycled uranium from commercial nuclear power plants. 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 steam generators (SGs), four motor driven pumps, four reactor inlet headers, four reactor outlet headers, one electrically heated pressurizer and all of the necessary interconnecting piping and valves. The system is arranged in a two loop, figure of eight configuration. The headers, SGs and pumps are all located above the reactor.

Examples may include a certain fuel loading or achievable burnup, unique core geometry or an innovative moderation technique.

Description of the safety systems

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 emergency core cooling system, with plant control and monitoring from the main control room. Group 2 comprises shutdown system 2, the emergency heat removal system and the containment system, with plant monitoring



Emergency core cooling system.

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.

Description of the turbine-generator systems

The turbine-generator, feedwater and condensate plant are completely located within the turbine building and are part of the balance of the 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.

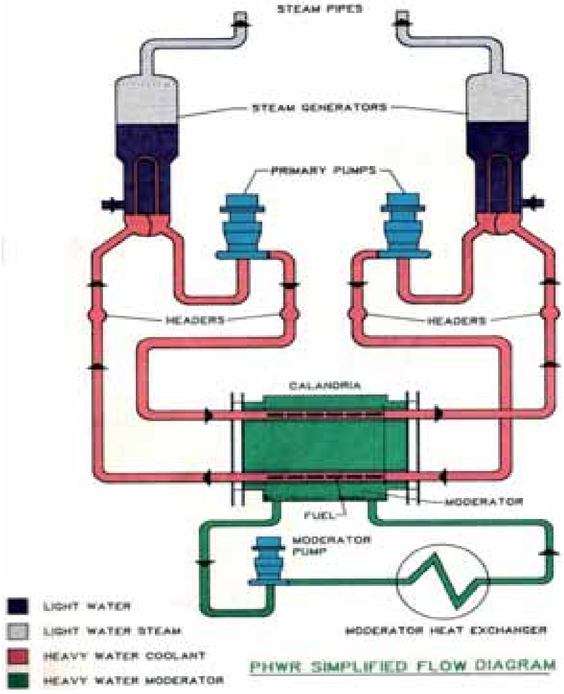
Electrical, and instrumentation and control systems

The process and reactor controls and safety functions are highly automated and provide information (i.e. plant parameters) to the operators using a combination of state of the art instrumentation and monitoring equipment. The main control room features extensive use of computer-driven, colour-graphic displays, which offer selective presentation of information in diagrammatic formats.

Deployment status and planned schedule

The Canadian Nuclear Safety Commission completed the Phase 2 pre-project design review in April 2012, concluding that there are no fundamental barriers to licensing the EC6 design in Canada [12].

PHWR-220 (NPCIL, India)



Reactor type:	Pressurized heavy water reactor
Electrical capacity:	236 MW(e)
Thermal capacity:	755 MW(th)
Coolant/moderator:	Heavy water (D_2O)
Primary circulation:	Forced circulation
System pressure:	8.5 MPa
Core outlet temperature:	293°C
Thermodynamic cycle:	Indirect Rankine
Fuel material:	UO_2
Fuel enrichment:	Natural uranium
Fuel cycle:	24 months
Reactivity control:	Absorber rods, booster rods and poison in the moderator
No. of safety trains:	2
Emergency safety systems:	Active and passive systems
Residual heat removal systems:	Active and passive systems
Design life:	40 years
Design status:	In operation
Seismic design:	0.2g
Predicted core damage frequency:	1E-5/reactor year
First date of completion:	1981 Rajasthan, India
Distinguishing features:	Proven design; indigenous Indian effort

Introduction

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

Description of the nuclear systems

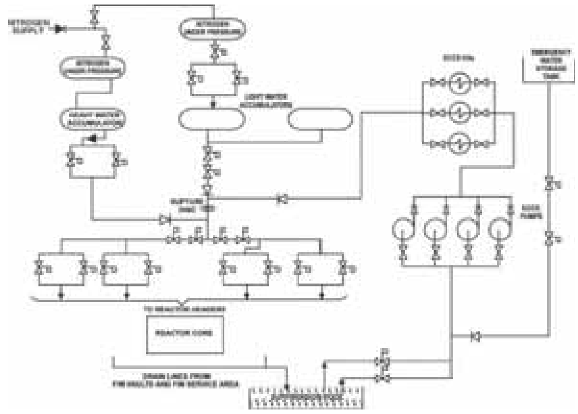
The PHWR uses heavy water as the moderator and coolant and natural uranium dioxide as fuel. The reactor consists of an integral assembly of two end shields and a calandria, with the latter submerged in the water filled vault.

During the residence period of the fuel in the reactor, about 1% of the uranium is burned. An increase in fuel burnup beyond 15 000 MW·d/t U using higher fissile content materials such as slightly enriched uranium, mixed oxide 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.

Primary coolant pumps (PCPs) circulate coolant through the reactor core. The primary heat transport (PHT) main circuit has four PCPs. The PCP is of a vertical, single stage type with a radial impeller located inside a volute casing that has an axial bottom entry suction and horizontal radial discharge. Each PCP is provided with three mechanical seals. Each of these mechanical seals can withstand full system pressure and, thus, provide reliable pressure boundary sealing. Four steam generators (SGs) are provided in the PHT system. The SGs are of a vertical mushroom type design with integral drum and feedwater preheaters.

Description of the safety concept

Indian PHWRs are designed and operated to achieve fundamental safety objectives in conformity with regulatory requirements of codes, guides and standards. 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. In a standard 220 MW(e) PHWR, the emergency core cooling system incorporates high pressure heavy water injection, intermediate pressure light water injection and low pressure long term recirculation. For standard PHWRs (220, 540 and 700 MW(e)), full double containment design has been adopted with primary containment of prestressed concrete and secondary containment of reinforced concrete.



Emergency core cooling system.

Electrical, and instrumentation and control systems

The control power supply system is divided into a main control power supply system and a supplementary control power supply system. The main control rooms are hybrid control rooms, wherein computer based operator information displays and parameter selection and settings facilities have been provided.

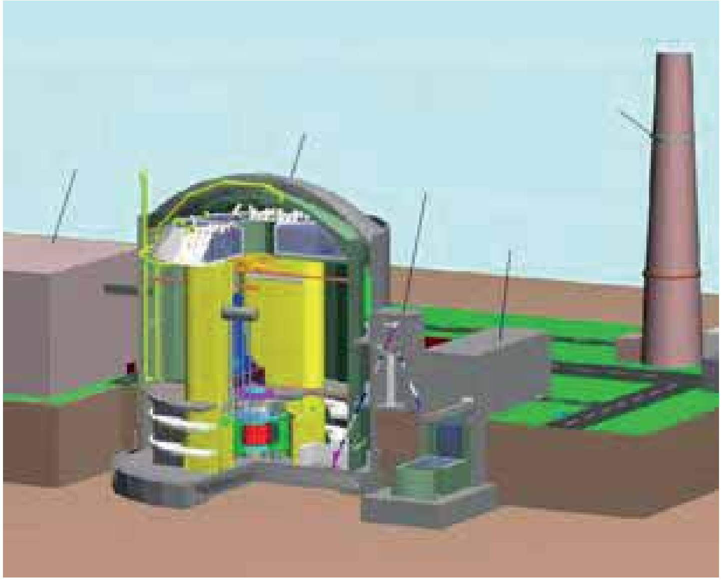
Description of the turbine-generator systems

Steam turbines for all of the 220 MW(e) PHWRs are configured with one single flow high pressure turbine and one double flow low pressure turbine tandem compounded and coupled to a two-pole generator.

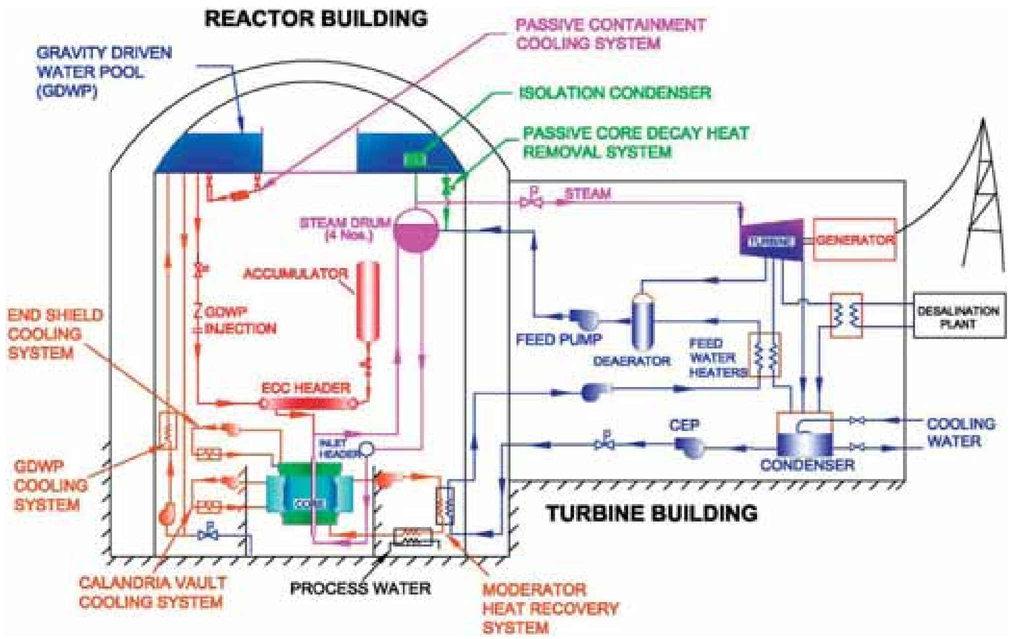
Deployment status and planned schedule

The PHWR reactors are currently deployed in India.

AHWR300-LEU (BARC, India)



Reactor type:	Pressure tube type heavy water moderated reactor
Electrical capacity:	304 MW(e)
Thermal capacity:	920 MW(th)
Coolant/moderator:	Light water/heavy water (D_2O)
Primary circulation:	Forced circulation
System pressure:	7.0 MPa
Core outlet temperature:	285°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	(Th, ^{233}U)–MOX (Th, Pu)–MOX
Fuel enrichment:	3.00–3.75% ^{233}U 2.50–4.00% Pu
Fuel cycle:	Closed cycle
Reactivity control:	Soluble boron/rod insertion
No. of safety trains:	4
Emergency safety systems:	Passive
Residual heat removal systems:	Passive
Design life:	100 years
Design status:	Basic design
Seismic design:	N/A
Predicted core damage frequency:	1E-8/reactor year
Planned deployment:	Site evaluation in progress
Distinguishing features:	Mixed oxide thorium closed fuel cycle; vertical cooling channels



General AHWR layout.

Introduction

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 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.

Description of the nuclear systems

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 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.

Description of the safety concept

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

Description of the turbine-generator systems

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, turbogenerator and auxiliaries, condensing system, condensate and feedwater heating system, steam dumping and relief systems and on-line full condensate flow purification.

Deployment status

Experiments are currently being carried out to test the various features of the reactor alongside site evaluations for future construction.

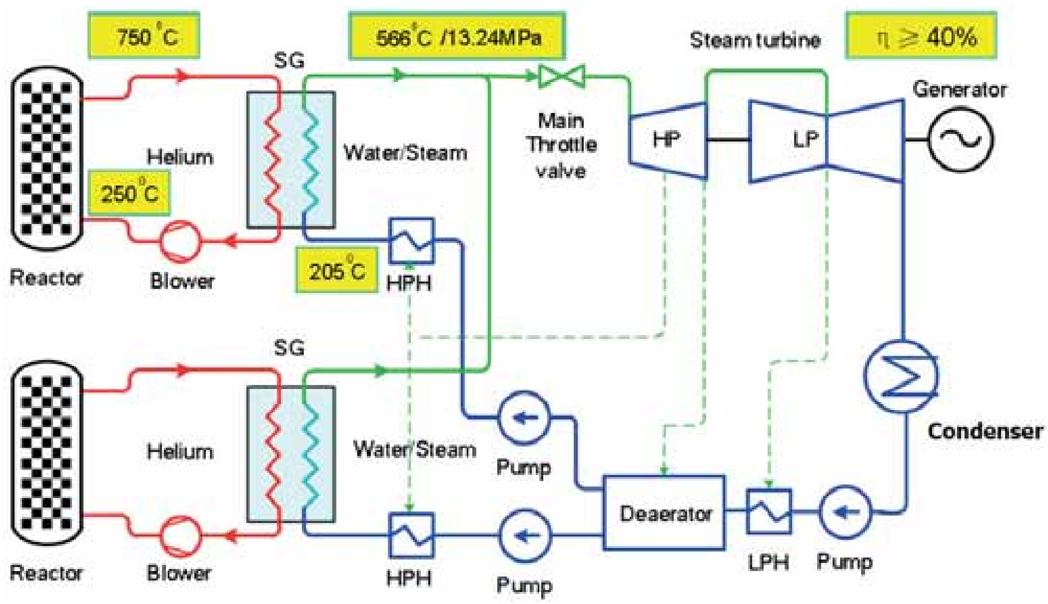
GAS

COOLED

REACTORS

HTR-PM (Tsinghua University, China)

Reactor type:	Pebble bed type reactor
Electrical capacity:	200 MW(e)
Thermal capacity:	500 MW(th)
Coolant/moderator:	Helium/graphite
Primary circulation:	Forced circulation
System pressure:	7 MPa
Core outlet temperature:	750°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	UO_2
Fuel enrichment:	8.5%
Fuel cycle:	1057 days
Reactivity control:	Rod insertion
No. of safety trains:	N/A
Emergency safety systems:	Passive
Residual heat removal systems:	Passive
Design life:	40 years
Design status:	Detailed design
Seismic design:	N/A
Predicted core damage frequency:	N/A
Planned deployment:	2013
Distinguishing features:	High efficiency electricity production with multiple reactors feeding one turbine-generator set



Plant layout.

Introduction

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 HTRs, 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, 13], when the HTR-PM project was launched.

Description of the nuclear systems

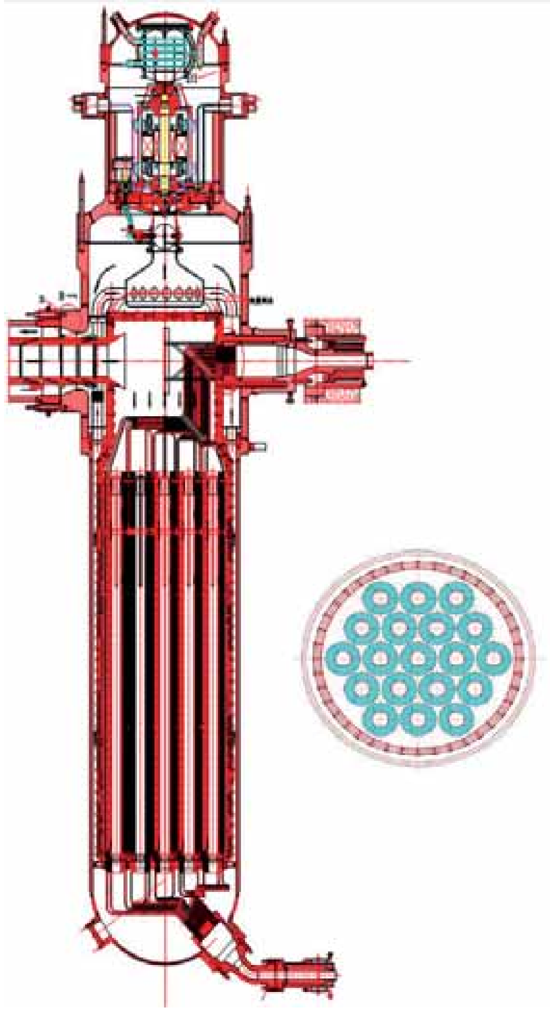
The HTR-PM utilizes the triple coated isotropic (TRISO) ceramic coated particle fuel element, which contains fuel kernels of 200–600 µm UO_2, UC_2 and UCO, but can also contain thorium or plutonium. The various layers of the TRISO fuel element enable it to tolerate more than 1600°C and retain fission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

The primary circuit consists of the reactor pressure vessel, the steam generator (SG) pressure vessel and the hot gas duct vessel connecting the two. 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. A helium test loop is currently being constructed to verify the design of the SG and ensure its design specifications [14].

Description of the safety concept

The HTR-PM incorporates the inherent safety principles of the modular HTGR, 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 of the 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.



Once-through helical steam generator.

There is no emergency core cooling system 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.

Deployment status

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 [13]. Construction began in 2009.

PBMR (PBMR Pty, South Africa)

Reactor type:	Pebble bed modular reactor
Electrical capacity:	164 MW(e)
Thermal capacity:	400 MW(th)
Coolant/moderator:	Helium/graphite
Primary circulation:	Forced circulation
System pressure:	9 MPa
Core outlet temperature:	900°C
Thermodynamic cycle:	Indirect Rankine
Fuel material:	TRISO UO_2 particles
Fuel enrichment:	9.6%
Fuel residence:	10 months
Reactivity control:	Rod insertion and absorber sphere channels
No. of safety trains:	N/A
Emergency safety systems:	N/A
Residual heat removal systems:	Reactor cavity cooling system
Design life:	60 years
Design status:	Basic design completed
Seismic design:	0.3g
Predicted core damage frequency:	N/A
Planned deployment:	PBMR Pty currently only maintaining intellectual property
Distinguishing features:	Post-shutdown decay heat removal achievable through conduction, natural convection and radiative heat transfer, where ultimate heat sink is concrete structure if reactor core cooling system fails

Introduction

The pebble bed modular reactor (PBMR) is an HTGR being designed and marketed by PBMR (Pty) Ltd.

The PBMR is designed for the electricity and process heat market. The reactor consists of triple coated isotropic (TRISO) fuel particles embedded within graphite pebbles, a helium coolant capable of achieving a core outlet temperature of 750°C, and a thermal output of 400 MW(th) with a power conversion efficiency of about 41%.

Description of the nuclear systems

The PBMR core is 3.7 m in diameter with a fixed central reflector of 2 m and an effective height of 11 m. The core contains about 452 000 fuel spheres, each with a loading of 9 g of uranium enriched at 9.6wt% ^{235}U that are being continuously cycled through the core. The graphite fuel spheres are 60 mm in diameter and contain TRISO coated fuel particles with a diameter of 0.92 mm. There are 24 partial length control rod positions in the side reflector serving as the reactivity control system, along with eight small absorber sphere systems positioned in the fixed central reflector as the reserve shutdown system which, when required, is filled with 1 cm diameter absorber spheres containing B_4C.

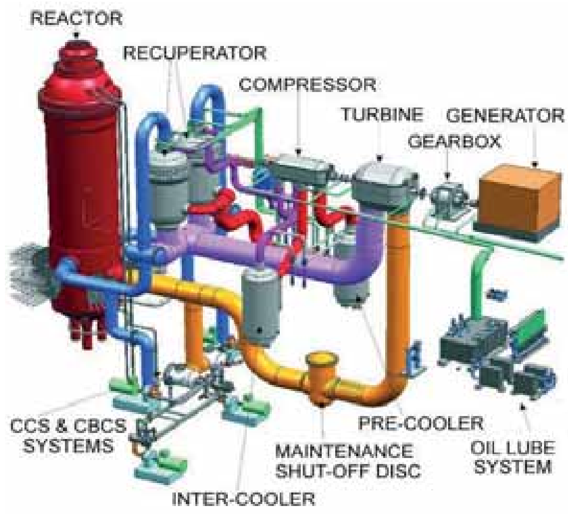
Description of the safety concept

The design ensures that the coated particles in the fuel spheres retain practically all radioactive fission products even during accident conditions. This feature removes the need for additional systems, electrical power and coolant to guarantee coolant circulation through the core after an accident.

The decay heat can be removed from the reactor by passive heat transport mechanisms such as heat conductivity, heat radiation and natural heat convection in combination with a heat exchanger that is placed in the citadel surrounding the reactor pressure vessel.

The shutdown systems are designed and placed in such a way that, when required, the absorber material can be inserted by means of gravity into borings in the graphite reflector.

Helium gas, used as the reactor coolant, is chemically inert and its density has essentially no influence on core reactivity. Loss of helium pressure does not lead to fuel failure.



Plant design.

Electrical, and instrumentation and control systems

The electrical supply system consists of a normal AC system based on the local standard of the client country and a DC system which has a battery backup with 24 h supply for important safety systems and 2 h for the main control system.

The instrumentation and control system has a digital plant control system to control all normal plant operational parameters, an equipment protection system to protect important equipment from damage, and the reactor protection system (RPS) which can override the other two systems. The main function of the RPS is to ensure safe reactor shutdown when certain limits are exceeded.

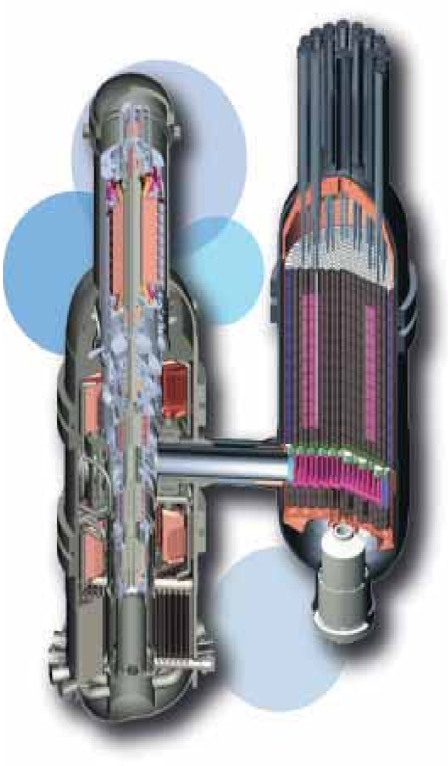
Description of the turbine-generator systems

The power conversion cycle is designed for a helium cooled direct Brayton cycle with a net electrical efficiency of about 41%. Various layouts are envisioned using a secondary water/steam cycle to provide electricity production along with process steam for industry applications or district heating.

Deployment status

The project to build a demonstration unit was abandoned in 2010 and the PBMR Company will exist until 2013 to care for and maintain the developed intellectual property.

GT-MHR (General Atomics, USA)



Reactor type:	High temperature gas cooled reactor
Electrical capacity:	150 MW(e)
Thermal capacity:	350 MW(th)
Coolant/moderator:	Helium/graphite
Primary circulation:	Forced circulation
System pressure:	6.39 MPa
Core outlet temperature:	750°C
Thermodynamic cycle:	Brayton cycle
Fuel material:	UCO
Fuel enrichment:	15.5%
Fuel cycle:	18 months
Reactivity control:	Rod insertion
No. of safety trains:	N/A
Emergency safety systems:	N/A
Residual heat removal systems:	N/A
Design life:	60 years
Design status:	Conceptual design
Seismic design:	N/A
Predicted core damage frequency:	N/A
Planned deployment:	N/A
Distinguishing features:	Efficient production of hydrogen by high temperature electrolysis or thermochemical water splitting

Introduction

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

Description of the nuclear systems

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.

Description of the safety concept

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 weapon production, as either fresh or as spent fuel.

Description of the turbine-generator systems

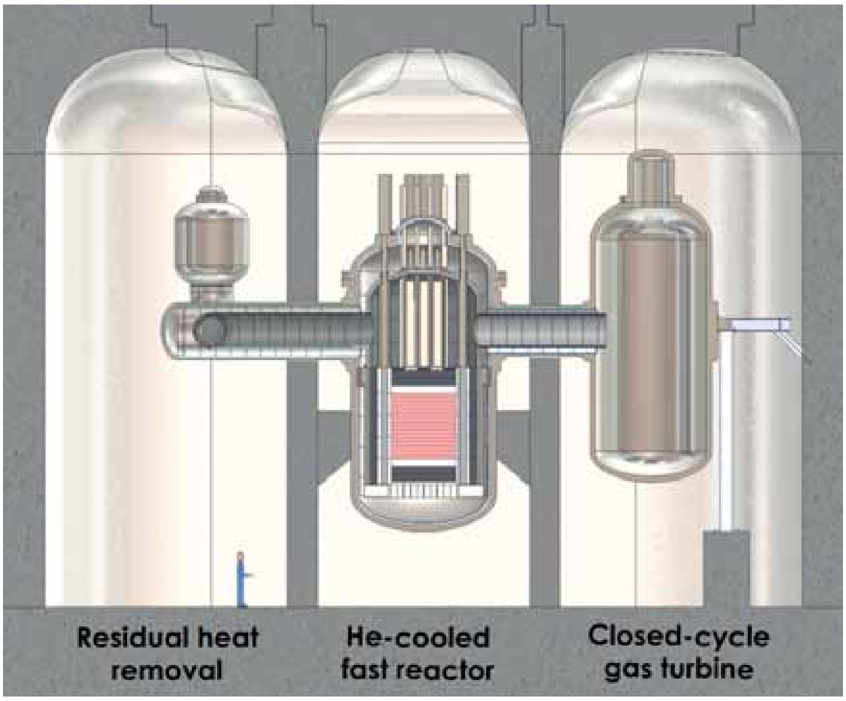
The GT-MHR direct Brayton cycle power conversion system contains a gas turbine, an electric generator and gas compressors. The use of the direct Brayton cycle results in a net plant efficiency of approximately 48%.

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.

Deployment status

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 [2].

EM² (General Atomics, USA)



Reactor type:	High temperature gas cooled fast reactor
Electrical capacity:	240 MW(e)
Thermal capacity:	500 MW(th)
Coolant:	Helium
Primary circulation:	Forced circulation
System pressure:	N/A
Core outlet temperature:	850°C
Thermodynamic cycle:	Direct Brayton cycle
Fuel material:	Used nuclear fuel
Fuel enrichment:	1% ^{235}U, 1% Pu, MA
Fuel cycle:	30 years
Reactivity control:	N/A
No. of safety trains:	N/A
Emergency safety systems:	N/A
Residual heat removal systems:	Passive
Design life:	30 years
Design status:	Conceptual design
Seismic design:	N/A
Predicted core damage frequency:	N/A
Planned deployment:	N/A
Distinguishing features:	Helium cooled fast reactor; reduces spent fuel inventories

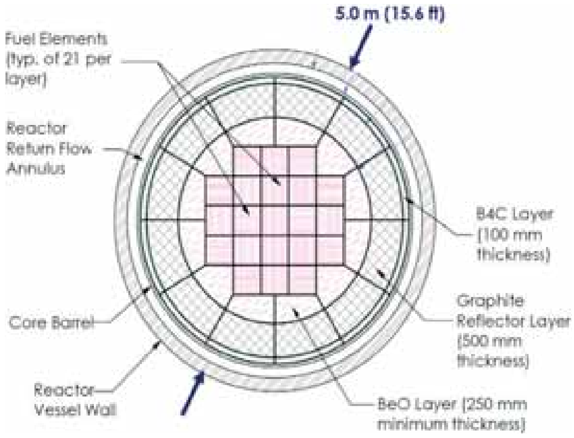
EM² reactor and gas turbine.

Introduction

The EM² 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.

Description of the nuclear systems

The reactor is designed to produce 500 MW(th) and 240 MW(e) based on a closed cycle gas turbine. The EM² 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 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.



Core layout.

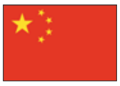
The core contains SiC–SiC clad porous UC plates arranged in a SiC–SiC assembly frame making a fuel assembly (FA). There are 21 FAs creating each layer and 17 layers stacked on top of each other, surrounded by a BeO layer, then a graphite reflector layer, and finally a B_4C layer, all sitting in the core barrel [15].

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 are removed.

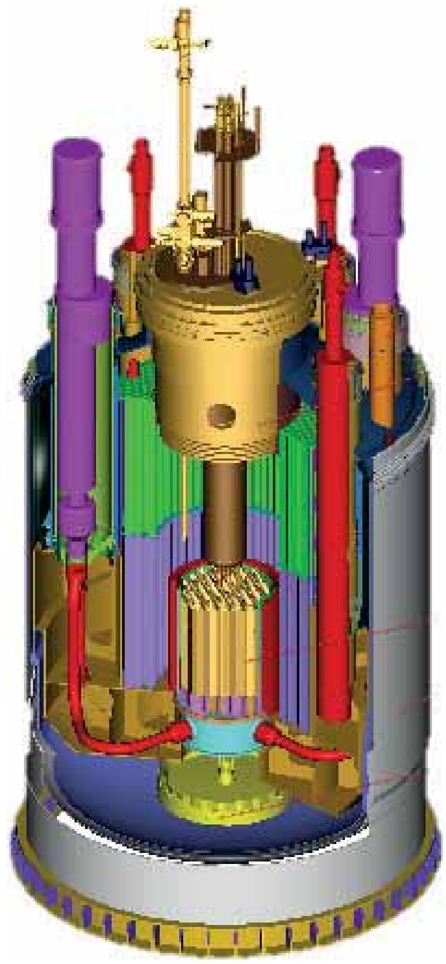
Description of the turbine-generator systems

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 [16].

LIQUID METAL COOLED REACTORS



CEFR (CNEIC, China)



Reactor type:	Liquid metal cooled fast reactor
Electrical capacity:	20 MW(e)
Thermal capacity:	65 MW(th)
Coolant:	Sodium
Primary circulation:	Forced circulation
System pressure:	Low pressure operation
Core outlet temperature:	530°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	$(Pu,U)–O_2$
Fuel enrichment:	19.6% ^{235}U
Fuel cycle:	N/A
Reactivity control:	Compensation, regulation and safety subassemblies
No. of safety trains:	N/A
Emergency safety systems:	Inherent features
Residual heat removal systems:	Passive
Design life:	30 years
Design status:	Operating
Seismic design:	N/A
Predicted core damage frequency:	N/A
Connection to grid:	July 2011
Distinguishing features:	Provides fast neutrons for irradiation testing; test bed for actual fast reactor conditions; model verification and experience needed for a commercial reactor

Introduction

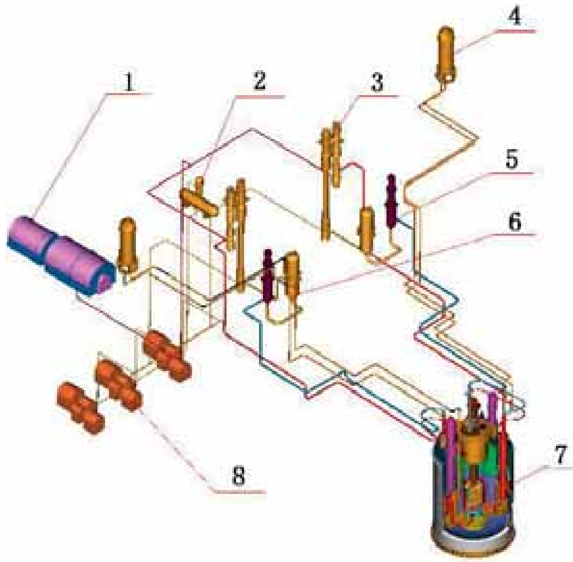
Based on two phases of fast reactor research and development targeting a 60 MW(th) experimental fast reactor, the China experimental fast reactor (CEFR) project was launched by the China Nuclear Energy Industry Corporation (CNEIC) in the framework of the national high-tech programme. The main objective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

Description of the nuclear systems

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 fuel assemblies, 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 516°C. The secondary circuit has two loops; each one is equipped with one secondary pump, two IHXs, an evaporator, a superheater, 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 superheater is 463.3°C. The tertiary water steam circuit provides 480°C and 14 MPa superheated steam to the turbine.



The CEFR main heat transfer system showing the (1) turbine-generator, (2) de-oxygen heater, (3) steam generator, (4) air cooler, (5) secondary pump, (6) buffer tank, (7) reactor block and (8) feedwater pumps [8].

Description of the safety concept

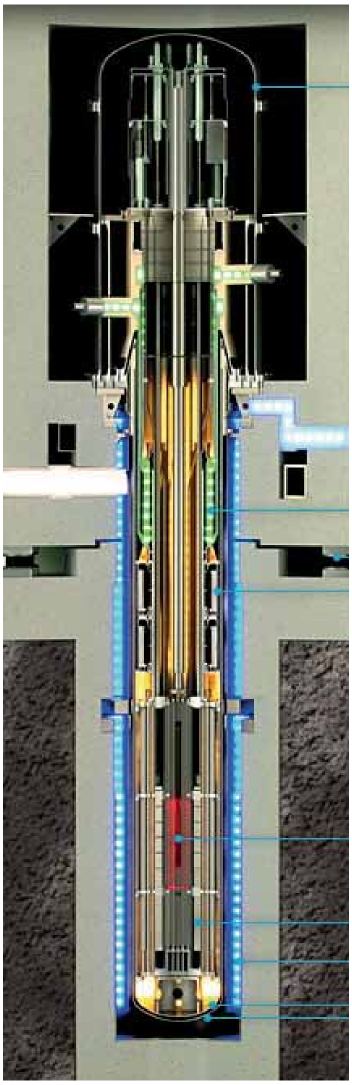
Sodium is contained within the reactor vessel, along with the core, the primary pumps and the IHXs, excluding the purification system. The CEFR is a small reactor that has greater heat inertia than many other pool reactors due to its larger primary sodium loading per megawatt(th). This provides long 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.

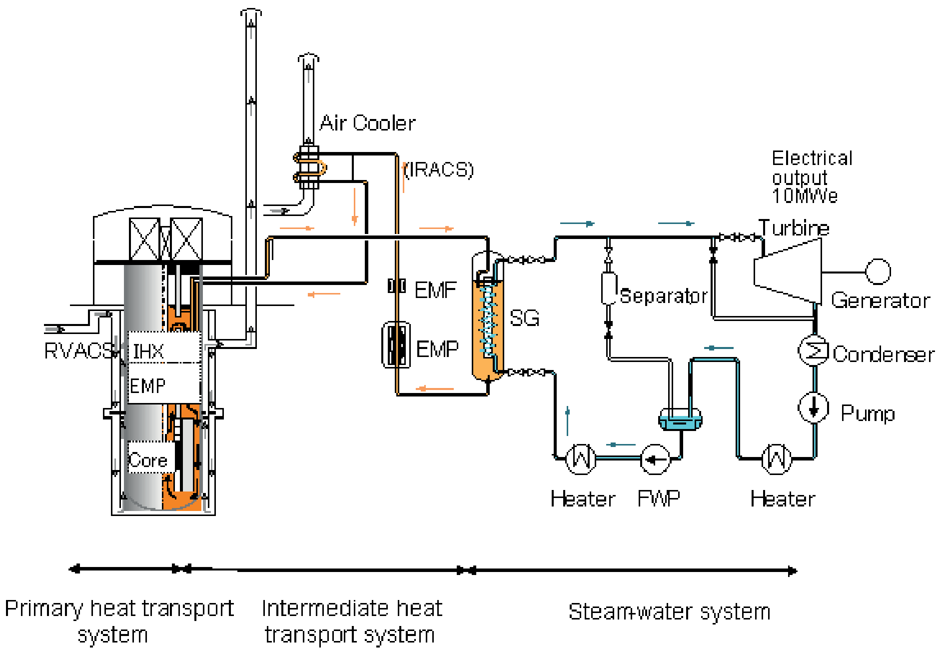
Deployment status

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 [17].

4S (Toshiba, Japan)



Reactor type:	Liquid metal cooled fast reactor
Electrical capacity:	10 MW(e)
Thermal capacity:	30 MW(th)
Coolant:	Sodium
Primary circulation:	Forced circulation
System pressure:	Non-pressurized
Core outlet temperature:	510°C
Thermodynamic cycle:	Indirect Rankine
Fuel material:	U–Zr alloy
Fuel enrichment:	<19%
Fuel cycle:	30 years
Reactivity control:	Soluble boron/rod insertion
No. of safety trains:	2
Emergency safety systems:	Active and passive
Residual heat removal systems:	Active and passive
Design life:	30 years
Design status:	In operation
Seismic design:	N/A
Predicted core damage frequency:	N/A
Planned deployment:	N/A
Distinguishing features:	Working with the city of Galena, Alaska, USA, for potential application



Simplified plant schematic.

Introduction

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 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.

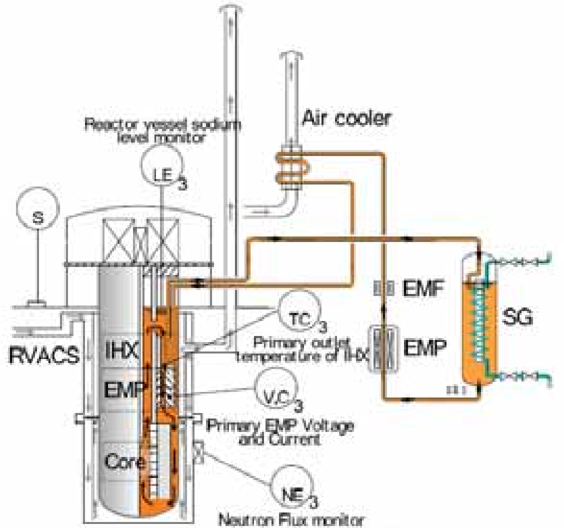
The 4S is a reactor the core of which has a lifetime of approximately thirty years. The movable reflector surrounding the core gradually moves, compensating for the burnup reactivity loss over the thirty year lifetime. The reactor power can be controlled by the water–steam system without affecting the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

Description of the nuclear systems

The reactor is an integral pool type, as all primary components are installed inside the reactor vessel. Major primary components are intermediate heat exchangers, 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.

Description of the safety concept

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.



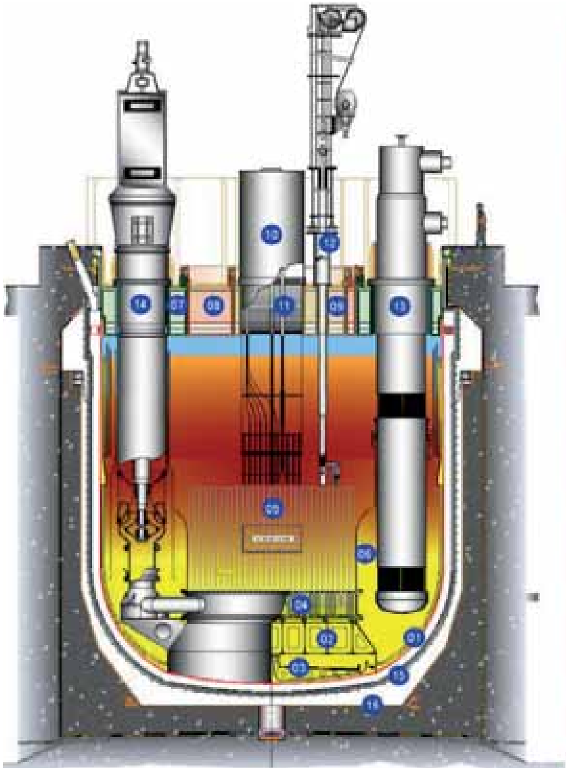
Reactor protection system sensors.

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 <20% by weight and a low plutonium content in the spent fuel (<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.

Deployment status

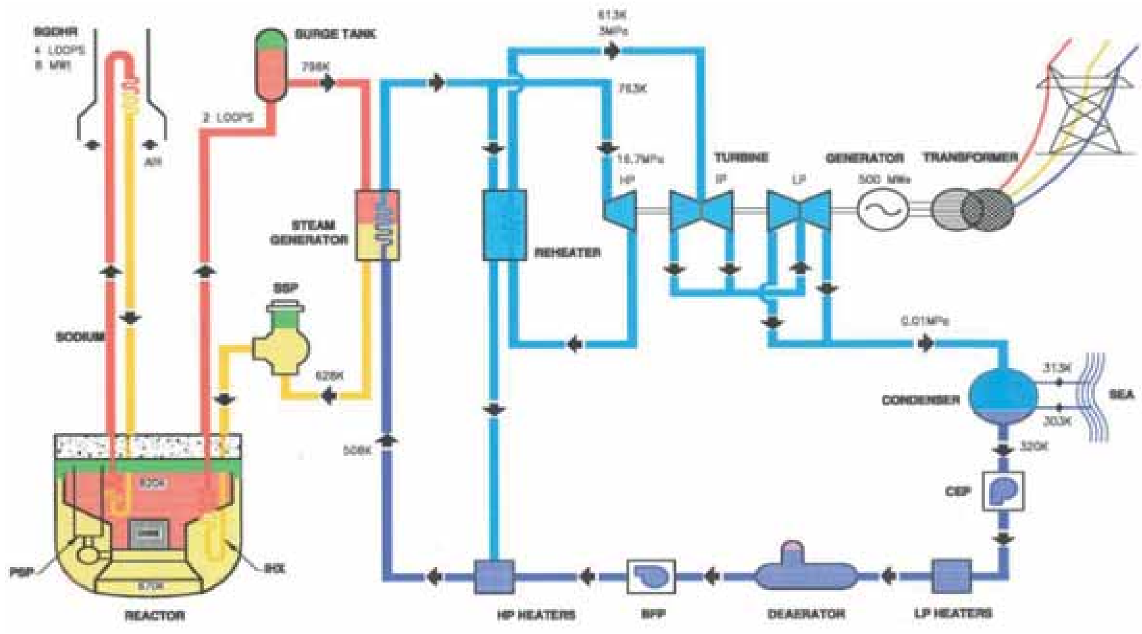
The 4S design is being developed at Toshiba and the Central Research Institute of the Electric Power Industry (CRIEPI) in Japan; Chubu Electric Power Company supported the initial phase of research and development relevant to the 4S. Research and development focusing on core, fuel and reflector technologies was conducted under the sponsorship of the Ministry of Education, Culture, Sports, Science and Technology in Japan, which included CRIEPI, the Japan Atomic Energy Research Institute, Osaka University and the University of Tokyo. The NRC is expecting design submittal in the third quarter of 2012 [18].

PFBR-500 (IGCAR, India)



01	MAIN VESSEL
02	CORE SUPPORT STRUCTURE
03	CORE CATCHER
04	GRID PLATE
05	CORE
06	INNER VESSEL
07	ROOF SLAB
08	LARGE ROTATABLE PLUG
09	SMALL ROTATABLE PLUG
10	CONTROL PLUG
11	CSRDM / DSRDM
12	TRANSFER ARM
13	IHX
14	PRIMARY SODIUM PUMP
15	SAFETY VESSEL
16	REACTOR VAULT

Reactor type:	Fast breeder reactor
Electrical capacity:	500 MW(e)
Thermal capacity:	1250 MW(th)
Coolant/moderator:	Sodium
Primary circulation:	Forced circulation
System pressure:	Low pressure operation
Core outlet temperature:	547°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	PuO_2–UO_2
Fuel enrichment:	2 enrichment zones
Fuel cycle:	6 months
Reactivity control:	Rod insertion
No. of safety trains:	4
Emergency safety systems:	Active
Residual heat removal systems:	Active primary, passive secondary and tertiary
Design life:	40 years
Design status:	Under construction
Seismic design:	N/A
Predicted core damage frequency:	N/A
Planned deployment:	2013
Distinguishing features:	First industrial scale fast breeder reactor in India



PFBR flow sheet [8].

Introduction

A 500 MW(e) prototype fast breeder reactor (PFBR) is being built by the Indira Gandhi Centre for Atomic Research at Kalpakkam; it will be the 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 mixed oxide fuel and depleted UO_2 as a blanket.

Description of the nuclear systems

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 intermediate heat exchangers (IHXs) 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 IHXs and sodium pumps, and axial leakages through the bottom fission product gas plenum. Mock-up shielding experiments were carried out to optimize design of this in-vessel shield.

There are eight integrated steam generator (SG) units, four per secondary loop where steam at 490°C 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 547°C and 397°C, respectively.

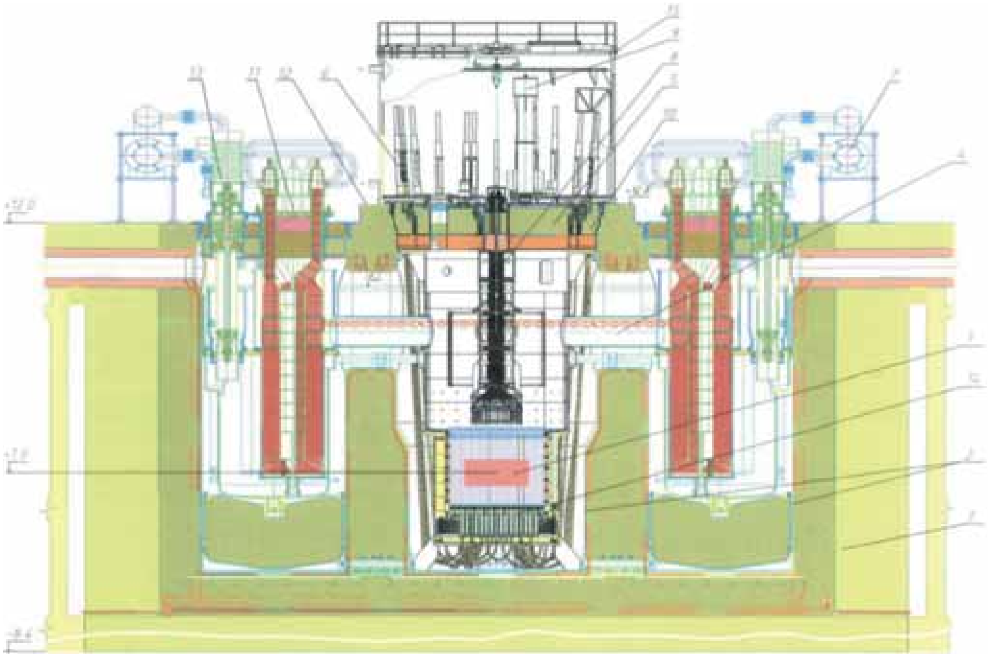
Description of the safety concept

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 sodium coolant in the entire primary system including the core and inner vessel is housed in 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.

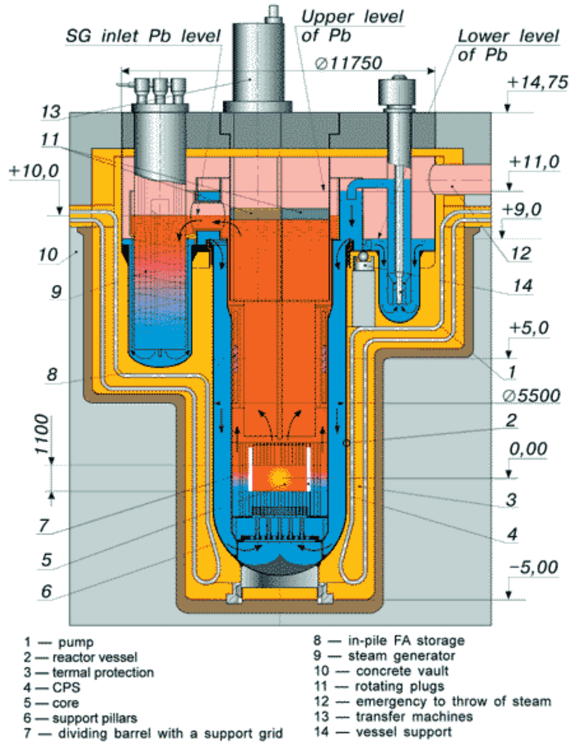
Deployment status

The PFBR is currently under construction and is expected to be commissioned in 2013 [19, 20].

BREST-OD-300 (RDIPE, Russian Federation)



Reactor type:	Liquid metal cooled fast reactor
Electrical capacity:	300 MW(e)
Thermal capacity:	700 MW(th)
Coolant:	Lead
Primary circulation:	Forced circulation
System pressure:	Low pressure operation
Core outlet temperature:	540°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	PuN–UN–MA
Fuel enrichment:	N/A
Fuel cycle:	12 months
Reactivity control:	N/A
No. of safety trains:	N/A
Emergency safety systems:	N/A
Residual heat removal systems:	N/A
Design life:	60 years
Design status:	Preliminary design
Seismic design:	VII–MSK 64
Predicted core damage frequency:	N/A
Planned deployment:	N/A
Distinguishing features:	High level of inherent safety due to natural properties of the fuel, core and cooling design



BREST-OD-300 primary systems.

Introduction

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.

Description of the nuclear systems

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 fuel assembly (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 that prevents overheating of the damaged FA.

BREST-OD-300 uses a mixed integral loop configuration of the primary circuit, as the steam generator (SG) and the 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.

Description of the safety concept

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 supercooling at the core inlet lead 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.

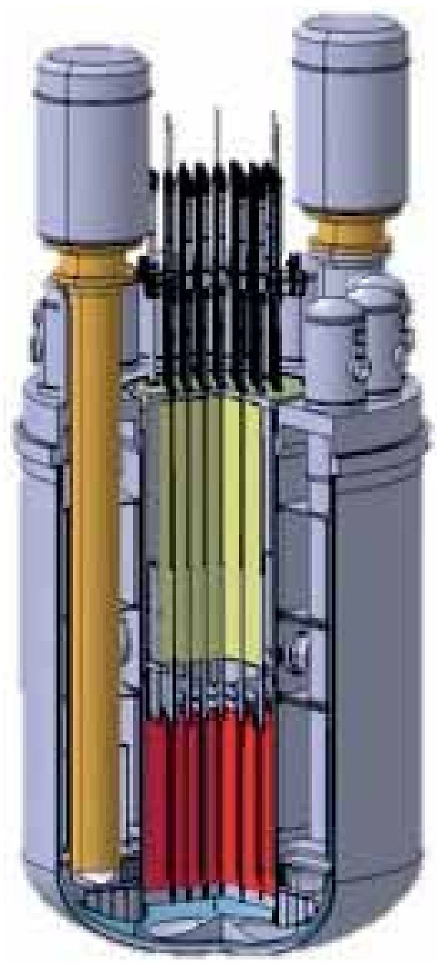
Description of the turbine-generator systems

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.

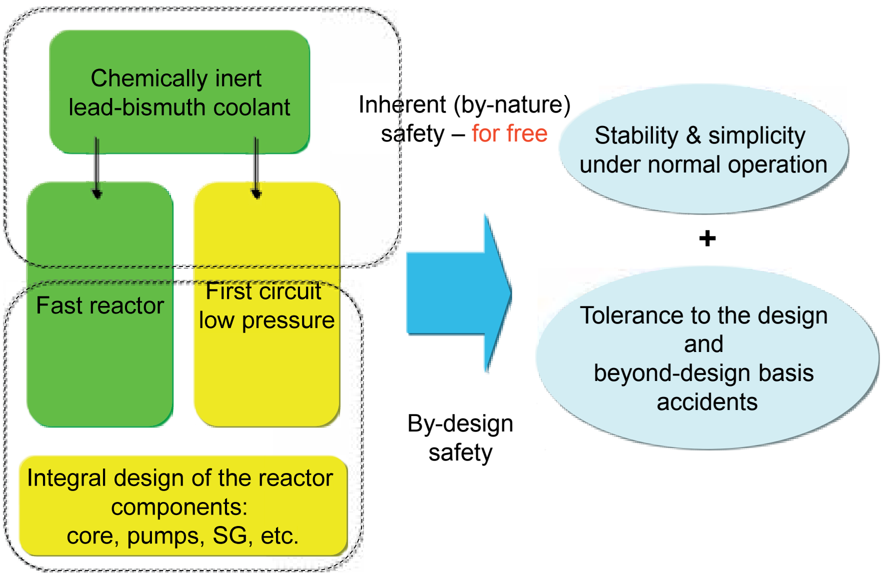
Deployment status

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 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 [17].

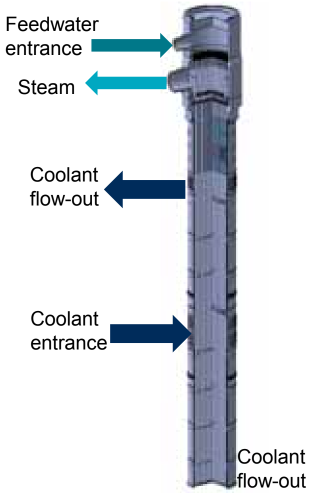
SVBR-100 (AKME Engineering, Russian Federation)



Reactor type:	Liquid metal cooled fast reactor
Electrical capacity:	101 MW(e)
Thermal capacity:	280 MW(th)
Coolant:	Lead–bismuth
Primary circulation:	Natural circulation
System pressure:	6.7 MPa
Core outlet temperature:	500°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	UO_2
Fuel enrichment:	<16.4%
Fuel cycle:	7–8 years
Reactivity control:	Control rod mechanism
No. of safety trains:	N/A
Emergency safety systems:	Passive
Residual heat removal systems:	Passive
Design life:	60 years
Design status:	Detailed design for construction in 2017
Seismic design:	0.5g
Predicted core damage frequency:	1E-8/reactor year
Planned deployment:	2021
Distinguishing features:	Closed nuclear fuel cycle with mixed oxide uranium plutonium fuel; operation in a fuel self-sufficient mode



Safety features.



SVBR-100 steam generator module.

Introduction

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.

Fuel cycle option

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 typical uranium oxide fuel, leading to a core breeding ratio (CBR) of 0.84; mixed oxide fuel can also be used, leading to a CBR just below one. 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 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.

Reactor coolant system

The entire primary equipment circuit of SVBR-100 is contained within a robust single reactor vessel; LBC valves and pipelines are all external. A protective enclosure surrounds the single unit reactor vessel. The reactor passes heat to a two circuit heat removal system and steam generator with a multiple circulation, secondary coolant system. Natural circulation of coolant in the reactor heat removal circuits is sufficient to passively cool down the reactor and prevent hazardous superheating of the core.

Description of the safety concept

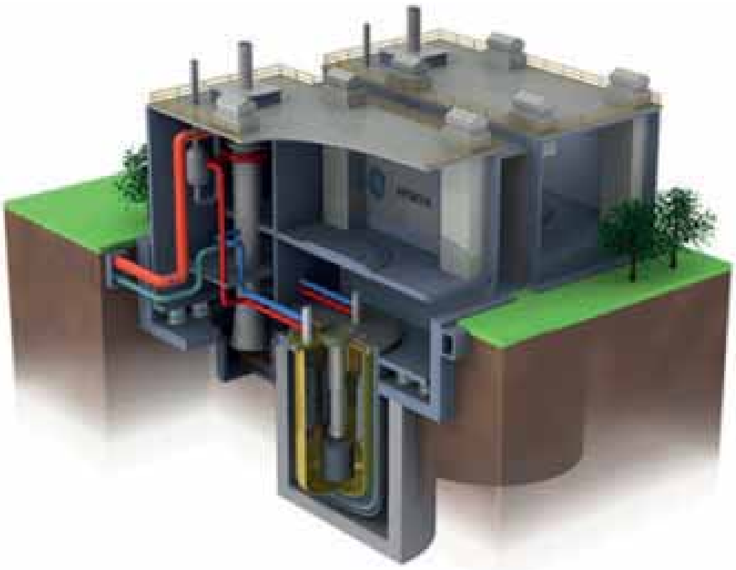
The combination of a fast reactor design with heavy metal coolant operating in an integral reactor layout ensures that the SVBR-100 reactor system meets IAEA international project safety levels for prevention of severe accidents and inherent safety, according to analysis and studies. The SVBR-100 design 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 disc membrane to prevent overpressurization and passive removal of residual heat in the event of a blackout.

Deployment status and planned schedule

The Rosatom Scientific and Technical Council convened on 15 June 2006 and approved the development of the technical design of the experimental industrial power unit based on the SVBR-100 [10, 17, 21, 22]. Siting licence works are underway; pilot plant specifications and key reactor and reactor core research and development works have begun. A complete reactor and power plant design is expected to be completed by 2013, along with a preliminary safety report. In 2013, a construction licence is also expected to be obtained. The trial unit is expected to be commissioned by 2017 at the Russian State Atomic Reactor Research Institute, Dimitrovgrad, in the region of Ulyanovsk [6].

PRISM (GE-Hitachi, USA)



Reactor type:	Liquid metal cooled fast breeder reactor
Electrical capacity:	311 MW(e)
Thermal capacity:	840 MW(th)
Coolant:	Sodium
Primary circulation:	Forced circulation
System pressure:	Low pressure operation
Core outlet temperature:	485°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	U–Pu–Zr
Fuel enrichment:	26% Pu,10% Zr
Fuel cycle:	18 months
Reactivity control:	Rod insertion
No. of safety trains:	N/A
Emergency safety systems:	Passive
Residual heat removal systems:	Passive reactor vessel auxiliary cooling system
Design life:	N/A
Design status:	Detailed design
Seismic design:	N/A
Predicted core damage frequency:	1E-6/reactor year
Planned deployment:	N/A
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

Introduction

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.

Description of the nuclear systems

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 lifetime for the fuel;
- To provide a 90 month lifetime for the blankets.

The reactor is designed to use a heterogeneous metal alloy core. The core consists of 42 fuel assemblies, 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 intermediate heat exchangers (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 upwards through the core and back into the hot pool.

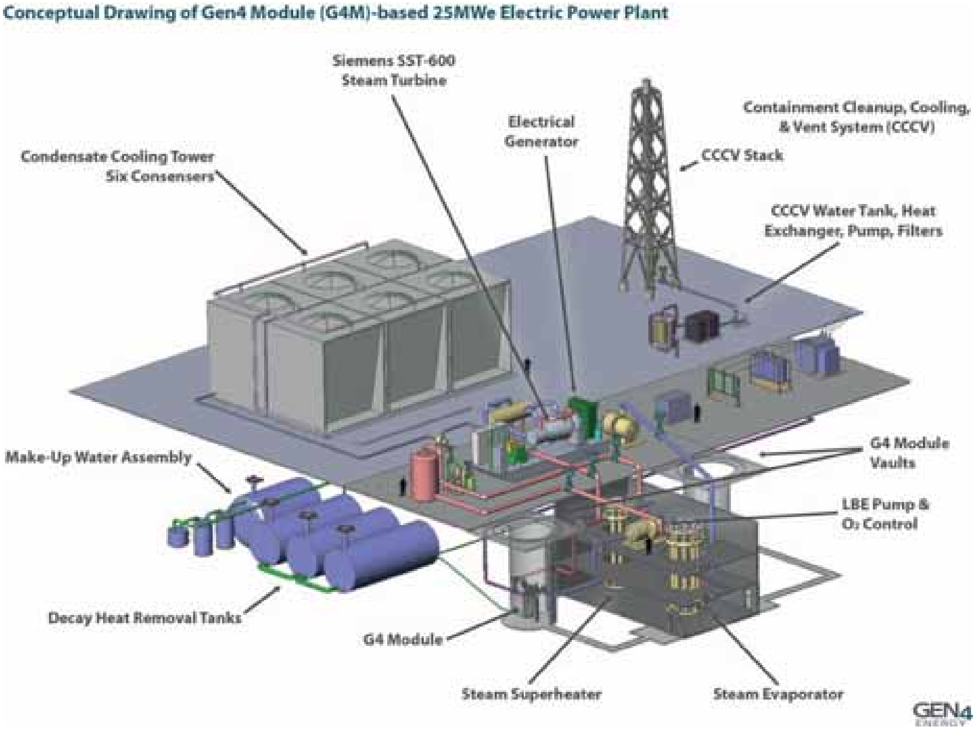
Description of the safety concept

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.

Description of the turbine-generator systems

A PRISM power block consists of three reactor modules, each with one steam generator (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 [23, 24].

G4M (Gen4 Energy Inc., USA)



Reactor type:	Liquid metal cooled reactor
Electrical capacity:	25 MW(e)
Thermal capacity:	70 MW(th)
Coolant:	Lead–bismuth
Primary circulation:	Forced circulation
System pressure:	N/A
Core outlet temperature:	500°C
Thermodynamic cycle:	Indirect Rankine cycle
Fuel material:	Uranium nitride
Fuel enrichment:	19.75%
Fuel cycle:	10 years
Reactivity control:	Rod insertion and B_4C ball insertion
No. of safety trains:	2
Emergency safety systems:	N/A
Residual heat removal systems:	Passive
Design life:	5–15 (nominal 10) years
Design status:	Conceptual design
Seismic design:	N/A
Predicted core damage frequency:	N/A
Planned deployment:	N/A
Distinguishing features:	Transportable factory fuelled design

Introduction

Founded in 2007 as Hyperion Power Generation Inc., Gen4 Energy was formed to develop the Gen4 Module (G4M), 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 [25].

Description of the nuclear systems

The reactor has been designed to deliver 70 MW of power over 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 B_4C 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%.

Description of the safety concept

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 G4M is driven by a set of design criteria that the designers believe are sufficient to ensure protection of the facility and its surroundings. These criteria are a sealed core, operational simplicity, minimal to no in-core movement, mechanical components and separation of power production and conversion operations.

During operational shutdowns, decay heat is removed from the G4M by two methods. The first method transfers heat from the core by natural circulation of coolants in the primary and secondary loops to the steam generators. The second removes heat by passive vaporization of water from the surface of the secondary containment vessel.

Deployment status

Gen4 Energy announced in April 2012 that they would not be pursuing the US Department of Energy's small modular reactor licensing support programme because they concluded that "use of well-known Light Water Reactor (LWR) technology of 45 to 300 MW intended for deployment in the USA had a much higher probability of success given the [Funding Opportunity Announcement's] stated maximum of two awards" [26].

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
CAREM	Integral pressurized water reactor	CNEA, Argentina	27	Site excavation completed
EC6	Pressurized heavy water reactor	AECL, Canada	740	Detailed design; CANDU 6 reference plants are in operation
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	180 × 2 modules	Pre-application interactions with the US NRC in July 2009; design certification application expected to be submitted in the fourth quarter of 2013

Reactor design	Reactor type	Designer, country	Capacity (MW(e))/ configuration	Design status
NuScale	Integral pressurized water reactor	NuScale Power Inc., USA	45 × 12 modules	NuScale plans to apply for design certification with the US NRC in 2013
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	Basic design
Flexblue	Subsea pressurized water reactor	DCNS, France	160 seabed anchored	Conceptual 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
G4M	Liquid metal cooled fast reactor	Gen4 Energy, USA	25 × N modules, single module or multimodule	Conceptual design
AHWR300-LEU	Pressure tube type heavy water moderated reactor	BARC, India	304	Basic design
FBNR	Integral pressurized water reactor	FURGS, Brazil	72	Concept description
SHELF	Pressurized water reactor	NIKIET, Russian Federation	6	Concept description

REFERENCES

[1] SMR vendors apply for government funds, World Nuclear News (22 May 2012).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Innovative Small and Medium Sized Reactor Designs 2005 — Reactors with Conventional Refuelling Schemes, IAEA-TECDOC-1485, IAEA, Vienna (2006).

[3] PAKISTAN ATOMIC ENERGY COMMISSION, Chashma Nuclear Power Plant-1 (CHASNUPP-1), PAEC, Islamabad (2011),
http://www.paec.gov.pk/chasnupp1

[4] JOINT STOCK COMPANY 'AFRIKANTOV EXPERIMENTAL DESIGN BUREAU FOR MECHANICAL ENGINEERING', Small Power Plants with ABV–6M Reactors, JSC 'OKBM Afrikantov', Nizhny Novgorod (1999),
http://www.okbm.nnov.ru/images/pdf/eng/abv_eng.pdf

[5] BELKIN, M., ROSATOMFLOT, The Traffic Regime on the Northern Sea Route Today & in the Future, Rosatomflot, Murmansk (2012),
http://rosatomflot.ru/rosatomflot.pdf

[6] JOINT STOCK COMPANY AFRIKANTOV OKB MECHANICAL ENGINEERING, Versatile Icebreaker, JSC Afrikantov OKBM, Nizhny Novgorod (2012),
http://www.okbm.nnov.ru/english/icebreaker

[7] KUZNETSOV, Y.N., GABARAEV, B.A., ROMENKOV, A.A., Co-generation Nuclear Power Plant with Innovative Simplified Passive Boiling Water Reactor VK-300, ICAPP'03, Cordoba, Spain (2003).

[8] KAZAROMPROM JSC, VBER-300 (2012),
http://www.kazatomprom.kz/en/pages/VBER-300

[9] Turbines installed on floating plant, World Nuclear News (10 Aug. 2011).

[10] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Small Reactor Designs without On-site Refuelling, IAEA-TECDOC-1536, IAEA, Vienna (2007).

[11] BABCOCK & WILCOX NUCLEAR ENERGY, Inc., Generation mPower™, B&W, Charlotte, NC (2011),
http://www.babcock.com/products/modular_nuclear/generation_mpower.html

[12] CANADIAN NUCLEAR SAFETY COMMISSION, Phase 2 Executive Summary: Pre-Project Design Review of Candu Energy Enhanced CANDU 6 Reactor — EC6, CNSC, Ottowa (2012).

[13] TSINGHUA UNIVERSITY INSTITUTE OF NUCLEAR AND NEW ENERGY TECHNOLOGY, HTR-PM, Beijing (2010),
http://www.tsinghua.edu.cn/publish/ineten/5695

[14] YULIANG, S., "Technology development, design and safety features of HTR-PM and its deployment scheme", presented at IAEA Workshop on Technology Assessment of Small and Medium-sized Reactors for Near Term Deployment (2011).

[15] RAWLS, J., "Implications for waste handling of the energy multiplier module", presented at the US Nuclear Waste Technical Review Board Full Board Mtg, Idaho Falls, ID (2010).

[16] GENERAL ATOMICS AND AFFILIATED COMPANIES, EM2™ Energy Multiplier Module, General Atomics, San Diego, CA (2010),
http://www.ga.com/energy/em2/

[17] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of Fast Reactor Research and Technology Development, Technical Reports Series No. 474, IAEA, Vienna (in preparation).

[18] NUCLEAR REGULATORY COMMISSION, Super-Safe, Small and Simple (4S), NRC, Washington, DC (1994), *http://www.nrc.gov/reactors/advanced/4s.html*

[19] India's first PFBR to go critical early 2013 (21 Jan. 2012), *http://zeenews.india.com/business/news/technology/india-s-first-pfbr-to-go-critical-early-2013_754072.html*

[20] SUBRAMANIAN, T.S., Prototype Fast Breeder Reactor 'has independent safety mechanisms', The Hindu (30 Apr. 2011).

[21] INTERNATIONAL ATOMIC ENERGY AGENCY, Power Reactors and Sub-critical Blanket Systems with Lead and Lead–bismuth as Coolant and/or Target Material, IAEA-TECDOC-1348, IAEA, Vienna (2003).

[22] AKME-ENGINEERING JSC, SVBR-100 (2012), *http://www.akmeengineering.com/svbr100.html*

[23] GE HITACHI NUCLEAR ENERGY, PRISM Technology Update, GEH, Wilmington, NC (2009), *http://www.ge-energy.com/content/multimedia/_files/downloads/dataform_2053733743_2809794.pdf*

[24] NUCLEAR REGULATORY COMMISSION, Preapplication Safety Evaluation Report for the Power Reactor Innovative Small Module (PRISM) Liquid–Metal Reactor, Rep. NUREG-1368, NRC, Washington, DC (1994).

[25] GEN4 ENERGY, Gen4 Energy, Inc., Denver, CO (2012), *http://www.gen4energy.com/about/*

[26] RUDIN, F., Gen4 Energy decides to withdraw its pursuit of the DOE SMR Funding Opportunity Announcement, Gen4 Energy, Denver (24 Apr. 2012), *http://www.gen4energy.com/news_item/gen4-energy-decides-to-withdraw-its-pursuit-of-the-doe-smr-funding-opportunity-announcement/*