Advances in Small Modular Reactor Technology Developments

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

IAEA

International Atomic Energy Agency

Atoms for Peace

ADVANCES IN SMALL MODULAR REACTOR TECHNOLOGY DEVELOPMENTS

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FOREWORD

The IAEA Department of Nuclear Energy devotes a number of its initiatives to support the development and deployment of small and medium-sized reactors (SMRs), recognizing their potential as options for enhancing the energy supply security both in expanding and embarking countries. Recently, the IAEA saw an increase in the participation of Member States in its programme for the technology development of SMRs. The driving forces in the development of such reactors are: meeting the need for flexible power generation for wider range of users and applications; replacing the ageing fossil fuel-fired power plants; enhancing safety performance through inherent and passive safety features; offering better economic affordability; suitability for non-electric applications; options for remote regions without established electricity grid infrastructures; and offering possibilities for synergetic energy systems that combine nuclear and alternate energy sources.

The trend in development has been towards design certification of small modular reactors, which are defined as advanced reactors that produce electric power up to 300 MW(e), designed to be built in factories and shipped to utilities for installation as demand arises. These new factory-built designs aim to reduce lengthy construction times while simultaneously increasing quality, thereby minimizing the financing costs associated with nowadays construction projects that span 5–8 years. SMR designs include water-cooled reactors, high temperature gas cooled reactors, as well as liquid metal cooled reactors with fast neutron spectrum. Some of SMRs are to be deployed as multiple-module power plants. Several countries are also pioneering in the development and application of transportable nuclear power plants, including floating and seabed-based SMRs. The distinct concepts of operations, staffing and security requirements, size of emergency planning zones (EPZs), licensing process, legal and regulatory framework are the main issues for the SMRs deployment. The projected timelines of readiness for deployment of SMRs generally range from the present to 2025–2030.

Member States, both those considering their first nuclear power plant and those with an existing nuclear power programme, are interested in information about advanced SMR designs and concepts, as well as new development trends. The IAEA Department of Nuclear Energy, which has been facilitating Member States in addressing common technologies and issues for SMRs and related fuel cycles in the past three decades, plays a prominent role in presenting international scientific forums and technical cooperation in this field for interested Member States.

The IAEA has been regularly publishing booklets on the status of SMR technology developments with the objective to provide Member States, including those considering initiating a nuclear power programme and those already having practical experience in nuclear power, a balance and objective overview of the status of SMR designs. This booklet is reporting the advances in global development of small modular reactor designs and technologies. This booklet covers only water-cooled and high temperature gas cooled small modular reactor designs and technologies. As for various small reactors with fast neutron spectrum, they are reported in a recent dedicated booklet on status of innovative fast reactor designs and concepts (see Annex III).

This booklet is intended as a supplement to the IAEA Advanced Reactor Information System (ARIS), which can be accessed at http://aris.iaea.org.

This publication was developed by Nuclear Power Technology Development Section, Division of Nuclear Power of the IAEA Department of Nuclear Energy in cooperation with Member States. The IAEA acknowledges the roles and contributions of A. Iunikova in coordinating the development of this booklet.

The IAEA officers responsible for this publication were H. Subki and F. Reitsma of the Division of Nuclear Power.

CONTENTS

INTRODUCTION

Small and medium sized reactors (SMRs) include a large variety of designs and technologies and in general, consist of:

- advanced SMRs, including modular reactors and integrated PWRs;
- innovative SMRs, including small-sized Gen-IV reactors with non-water coolant/ moderator;
- converted or modified SMRs, including barge mounted floating NPP and seabed-based reactors;
- conventional SMRs, those of Gen-II technologies and still being deployed.

Advanced SMRs having an equivalent electric power of less than 700 MW(e) or even less than 300 MW(e) are part of a new generation of nuclear power plant designs being developed to provide a flexible, cost-effective energy for various applications. Advanced SMR designs include water-cooled reactors, high-temperature gas cooled reactors, as well as liquid-metal cooled reactors with fast neutron spectrum. The trend of development has been towards design certification of small modular reactors, which are defined as advanced reactors that produce equivalent electric power less than 300 MW(e) designed to be built in factories and shipped to utilities for installation as demand arises. The SMR systems adopt modularization, by which the structures, systems and components are shop-fabricated then shipped and assembled onsite, thus the construction time for SMRs can be substantially reduced. Some of the SMRs are to be deployed as multiple-module power plants allowing utilities to add additional reactor and power conversion modules as demand for local power increases.

Advanced SMRs will use different approaches from large reactors for achieving a high level of safety and reliability in their systems, structures, components, and that will be the result of a complex interaction between design, operation, material and human factors. Interest in SMRs continues to grow as an option for future power generation and energy security. However, the first phase of advanced SMR deployments will have to ultimately demonstrate high levels of plant safety and reliability, and prove their economics in order for further commercialization to be feasible. This situation is an issue for the technology developers since building the first plants will be relatively expensive and the ‘n-th of a kind’ costs can only be confirmed after the first demonstration plants are deployed. As a result, new private-public partnership arrangements may be needed to support the first phase of advanced SMR deployments. These plants would have greater automation but will still rely on human interaction for supervision, system management, and operational decisions because operators are still regarded as the last line of defence if failures in automated protective measures occur.

Currently there are more than 45 SMR designs under development for different application issues. In 2014, four reactors in the SMR category are under construction in Argentina (CAREM 25, an industrial prototype), the Russian Federation (KLT-40S, a barge mounted floating power unit, RITM-200 for nuclear icebreakers) and in China (HTR-PM, an industrial demonstration plant). The distinct concepts of operation, licensing process, legal and regulatory framework are the main issues for the SMRs deployment. The projected timelines of readiness for deployment of SMR designs generally range from the present to 2025–2030.

The Central Argentina de Elementos Modulares (CAREM) reactor is a small, integral type pressurized light water reactor (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-25 prototype was completed at the end of August 2012 and construction has begun and contracts with different Argentinean stakeholders for manufacturing of the components have already been signed. In July 2012, the Korean Nuclear Safety and Security Commission issued the Standard Design Approval for the 100 MW(e) System Integrated Modular Advanced Reactor (SMART) – the first integrated PWR received certification. The China National Nuclear Corporation is developing the ACP100 design and will submit its preliminary safety analysis report to the National Nuclear Safety Administration (NNSA) in 2014 for construction in 2016.

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 construction is to be completed by the end of 2016 and expected electricity production is by 2017. The ABV-6M with an electrical output of 8.6 MW(e) is a nuclear steam generating plant using natural circulation for its integral reactor coolant system. Its development is at the final design stage. The RITM-200, an integral reactor with forced circulation for universal nuclear icebreakers, is designed to generate 50 MW(e). Two reactor plants of RITM-200 are being manufactured for the first multipurpose icebreaker aiming for complete delivery in 2016. A follow up deliveries of reactors for two consequent nuclear icebreakers will be in 2017 and 2018. The VK-300 is a 250 MW(e) boiling water reactor (BWR) that operates with natural circulation and employs passive residual heat removal systems (RHRSs). Research and development activities are currently under way for further validation and actualization of the design approach adopted in the VK-300 design. The VBER-300 is a 325 MW(e) pressurized water reactor (PWR) conceptual design that can also serve as a power source for floating nuclear power plants. The N.A. Dollezhal Research and Development Institute of Power Engineering (NIKIET) is designing the UNITHERM to generate 6.6 MW(e), based on design experience in marine nuclear installations, and the SHELF PWR, a 6 MW(e) underwater, remotely operated power source.

Four integral pressurized water SMRs are under development in the USA: Babcock & Wilcox's mPower, NuScale, SMR-160 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 short term. NuScale Power envisages a nuclear power plant made up of twelve modules producing more than 45 MW(e) and has a target commercial operation in 2023 for the first plant that is to be built in Idaho. The design certification application of NuScale to the NRC is expected in the second half of 2016. 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. The SMR-160 design generates power of 160 MW(e) adopting passive safety features and its conceptual design is to be completed in 2015.

One of the research and development activities on SMRs in academic domain is performed by the Politecnico di Milano (POLIMI) in Italy and universities in Croatia and Japan that are continuing the development of the International Reactor Innovative and Secure (IRIS), previously lead by the Westinghouse consortium. IRIS is an integral PWR design with an electrical capacity of 335 MW(e). In France, the Flexblue design is a small seabed nuclear power plant with an output of 160 MW(e) with a target deployment by 2025. In Japan, at least two water-cooled SMR designs are under development. The DMS design (Double MS: Modular Simplified and Medium Small Reactor) is a small-sized boiling water reactor (BWR) which generates 300 MW(e). The Integrated Modular Water Reactor (IMR) is a medium sized power reactor producing electricity of 350 MW(e). Validation testing, research and development for components and design methods, and basic design development are required before licensing. For heavy-water cooled SMRs, India has been developing the advanced heavy water reactor (AHWR300-LEU) to provide 304 MW(e). The design incorporates vertical pressure tubes, low enriched uranium and thorium fuel, and passive safety features; it is currently in the basic design stage.

In the innovative reactors arena, high temperature gas cooled reactors (HTGRs) have an inherent safety benefit whereby extremely high temperature values can be tolerated without fuel damage, and this also provides high temperature heat (≥750°C) that can be utilized for a variety of industrial applications as well as for cogeneration. This means that HTGRs can contribute to the total energy market in addition to electricity generation. As generally applied for water-cooled SMRs, the smaller size and simplified design (with a small number of safety systems) also make HTGRs potentially attractive to Member States with small electricity grids. Furthermore, the significant process heat options far exceed the capability of light water reactors and hence make them even more attractive. The construction of the High Temperature Reactor–Pebble Bed Module (HTR-PM) industrial demonstration power plant in China will also make such technology available for near term deployment. The HTR-PM is a unique twin nuclear steam supply system feeding a single 200 MW(e) superheated steam turbine generator. Construction has started in December 2012 with first expected operation by the end of 2017.

Japan Atomic Energy Agency (JAEA) has several conceptual designs based on the experience and development work related to the HTTR test reactor. Presented here is the GTHTR300 (Gas turbine High Temperature Reactor 300 MW(e)), a multipurpose, inherently-safe and site-flexible small modular reactor. This reactor is under development for commercialization in 2020s. The development of HTGR technologies in Russia includes the project to develop the 285 MW(e) Gas Turbine-Modular Helium Reactor (GT-MHR) for electricity production but recently also the diversification of the nuclear power applications for industrial purposes, such as the MHR-T reactor/hydrogen production complex that use 4x600 MW(th) modules. The GT-MHR is a Russian Federation – USA jointly funded project, originally aimed at solving one of the most important tasks in the area of non-proliferation; the disposition of weapons-grade plutonium. Finally, the MHR-100 prismatic modular helium reactor design of 215 MW(th) is used in multiple configurations for electricity and cogeneration.

The South African developed Pebble Bed Modular Reactor (PBMR-400) design which can produce electricity (165 MW(e)) at high efficiency via direct Brayton cycle employing a helium gas turbine. The unique fixed central column design allows the larger thermal power in a pebble bed design while retaining its inherent safety characteristics for decay heat removal with passive only means even under the most severe conditions. The design information is secured and maintained. Also in South Africa the Steenkampskraal Thorium Limited (STL) company has finished a conceptual design of the HTMR-100 pebble bed design and plans to be able to use a range of uranium and plutonium-thorium coated particle pebble fuels in a once through-then-out (single pass) cycle. The HTMR-100 is designed to generate 35 MW(e).

In USA, the NGNP Industry alliance has selected the AREVA 272 MW(e) SC–HTGR a prismatic block design for commercialization and the preparation for pre-licensing application have started. The design is based on the AREVA's ANTARES concept but coupled to two steam generators and allows for cogeneration. A privately owned and funded initiative in the USA, called X-energy, is pursuing the Xe-100 small-sized pebble bed reactor producing 35 MW(e). A major aim of the design is to improve the economics through system simplification, component modularization, reduction of construction time and high plant availability brought about by continuous fuelling.

The fast neutron spectrum high temperature gas cooled SMR designs are reported in another dedicated booklet on status of innovative fast reactor designs and concepts (see Annex III).

In this booklet, all SMR designs are presented from the viewpoints of the motivation of their development, development milestones, target applications, specific design and safety features, fuel characteristics and fuel supply issues and finally their licensing and certification status. Not all small reactor designs presented in this booklet could be categorized as small modular reactors, as some of them strongly rely on proven technologies of operating large capacity reactors. They are presented in this booklet for reason of completeness and designers foresee certain niche markets for their products. The brief yet informative technical description and major technical parameters including the predictions of core damage frequencies (CDFs) were provided by the design organizations without validation or verification by the IAEA. All figures, illustrations and diagrams were also provided by the design organizations.

WATER

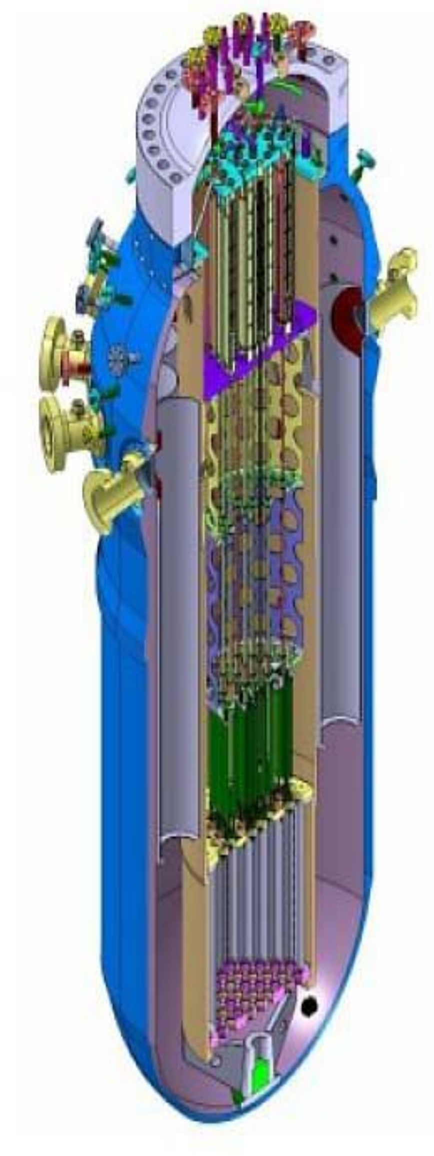
COOLED

REACTORS

CAREM-25 (CNEA, Argentina)

Introduction

Central Argentina de Elementos Modulares (CAREM-25) is a national SMR development project based on LWR technology coordinated by the Argentina National Atomic Energy Commission (CNEA) in collaboration with leading nuclear companies in Argentina with the purpose to develop, design and construct innovative small nuclear power plants with high economic competitiveness and high level of safety. CAREM-25 is deployed as a prototype to validate the innovations for future commercial version of CAREM that will generate an electric output of 150-300 MW(e). CAREM-25 is an integral type PWR based on indirect steam cycle with distinctive features that simplify the design and support the objective of achieving a higher level of safety. Some of the design characteristics of CAREM-25 are: integrated primary cooling system, in-vessel hydraulic control rod drive mechanisms and safety systems relying on passive features. Coolant flow in the primary reactor system is done by natural circulation. CAREM-25 reactor was developed using domestic technology, at least 70% of the components and related services were sourced from Argentine companies.



Reactor System Configuration of CAREM-25
(Courtesy of CNEA, with permission)

Development Milestones

1984	CAREM concept was presented in Lima, Peru, during the IAEA Conference on SMRs and was one of the first new generation reactor designs. CNEA officially launched the CAREM Project
2001-02	The design was evaluated on Generation IV International Forum and was selected in the near term development group
2006	Argentina Nuclear Reactivation Plan listed the CAREM-25 project among priorities of national nuclear development
2009	CNEA submitted its preliminary Safety Analysis Report (PSAR) for CAREM-25 to the ARN. Announcement was made that Formosa province was selected to host the CAREM
2011	Start-up of a high pressure and high temperature loop for testing the innovative hydraulic control rod drive mechanism (CAPEM)
2011	Site excavation work began; Contracts and agreements among stakeholders were discussed
2012	Civil engineering works
2014	8 February, formal start of construction
2018	First fuel load

Target Applications

CAREM-25 is designed as an energy source for electricity supply of regions with small demands. It can also support seawater desalination processes to supply water and energy to coastal sites.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	CNEA
Country of Origin:	Argentina
Reactor Type:	Integral PWR
Electrical Capacity (MW(e)):	27
Thermal Capacity (MW(th)):	100
Expected Capacity Factor (%):	> 90
Design Life (years):	60
Plant Footprint (m^2):	N/A
Coolant/Moderator:	Light water
Primary Circulation:	Natural circulation
System Pressure (MPa):	12.25
Main Reactivity Control Mechanism:	Control Rod Driving Mechanism (CRDM) only
RPV Height (m):	11
RPV Diameter (m):	3.2
Coolant Temperature, Core Outlet (°C):	326
Coolant Temperature, Core Inlet (°C):	284
Integral Design:	Yes
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	No
Low-Temp Process Heat:	Yes
Cogeneration Capability:	Possible
Design Configured for Process Heat Applications:	No
Passive Safety Features:	Yes
Active Safety Features:	Yes
Fuel Type/Assembly Array:	UO_2 pellet/hexagonal
Fuel Active Length (m):	1.4
Number of Fuel Assemblies:	61
Fuel Enrichment (%):	3.1 (prototype)
Fuel Burnup (GWd/ton):	24 (prototype)
Fuel Cycle (months):	14 (prototype)
Number of Safety Trains:	2
Emergency Safety Systems:	Passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	not available
Distinguishing Features:	Core heat removal by natural circulation Pressure suppression containment
Modules per Plant:	1
Estimated Construction Schedule (months):	~36
Seismic Design (g):	0.25
Predicted Core Damage Frequency (per reactor year):	10^{-07}
Design Status:	Under construction

Specific Design Features

Through the integrated design approach, the pressurizer, the control rod driving mechanism (CRDM) and twelve mini-helical once-through steam generators (SGs) are installed inside the reactor pressure vessel (RPV), i.e. in the same compartment with the reactor core. The location of the SGs 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 counter-current flow.

Due to self-pressurization, temperatures values of the core outlet, the pressuriser and the dome are very close to the saturation temperature. Under all operating conditions, this has proved to be sufficient to guarantee the required performance 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 possible with minimal control rod motion. The size of the coolant nozzles into the RPV is limited to 12 coaxial cross tubes within each nozzle so as to separate hydraulic connections for feedwater inlets and steam outlets through the vertical steam generators. This eliminates the probability of large break LOCA. The secondary piping system is similar to those of PWR. Electrically driven pumps are employed to provide pressurized feedwater to the inlet of the 12 steam generators through hydraulic connections within the steam nozzles.

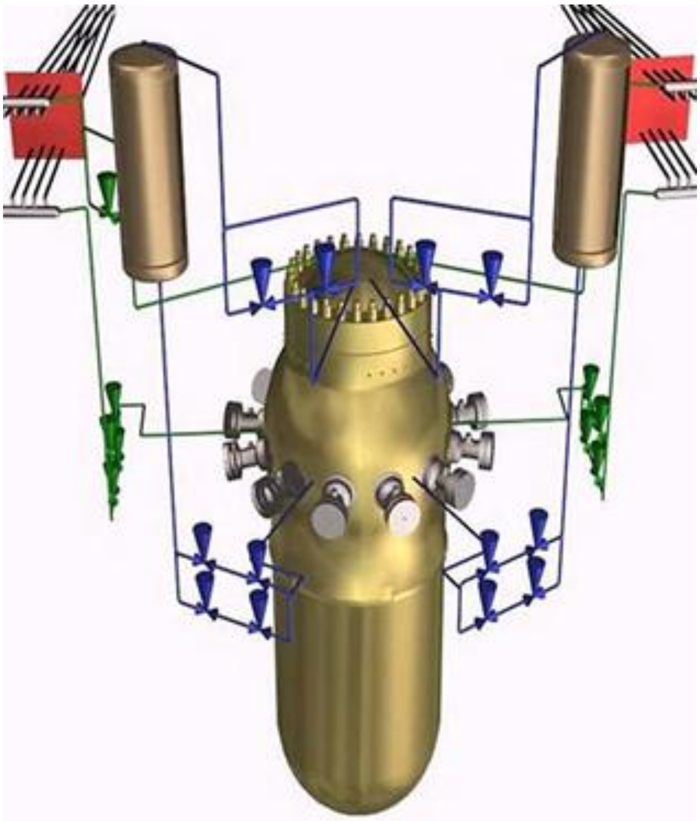
Safety Features

The defence in depth (DID) concept is based on Western European Nuclear Regulators Association (WENRA) proposal and include clarification on multiple failure events, severe accidents, independence between levels, the use of the SCRAM system in some DID Level #2 events and the containment in all the Protection Levels. The DID concept has been internalized in CAREM-25, in order to improve safety significantly, compared with the current NPP designs. Many intrinsic characteristics contribute to the preclusion of typical LWR initiating events, such as large and medium loss of coolant accidents (LOCA), loss of flow accidents, boron dilution and control rod ejection. CAREM-25 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. The safety systems are duplicated to fulfil the redundancy criteria, and the shutdown system is diversified to fulfil regulatory requirements.

The safety system of CAREM-25 consists of two reactor protection systems (RPS), two shutdown systems, passive residual heat removal system (PRHRS), safety valves, low pressure injection system, depressurization system and containment of pressure suppression type.

For the grace period (36hr), core decay heat removal can ensure safe core temperature due to availability of PRHRS in the case of loss of heat sink or station black-out (SBO). The PRHRS are heat exchangers formed by parallel horizontal U-tubes (condensers) coupled to common headers. A set of headers is connected to the RPV steam dome, while another set (condensate return line) is coupled with the RPV at the inlet of the primary system side of the SG. Though natural circulation, the design provides core decay heat removal, transferring it to dedicated pools inside the containment and then to the suppression pool. In CAREM-25, SBO is classified as a design basis event. Despite the low frequency of core meltdown, severe accident prevention is considered for grace period prolongation under the hypothesis of SBO longer than 72 hours, simple systems supported by fire extinguishing system or external pumps and containment protection. Regarding severe accident mitigation, provisions are considered for Hydrogen control and for RPV lower head cooling for in-vessel corium retention.

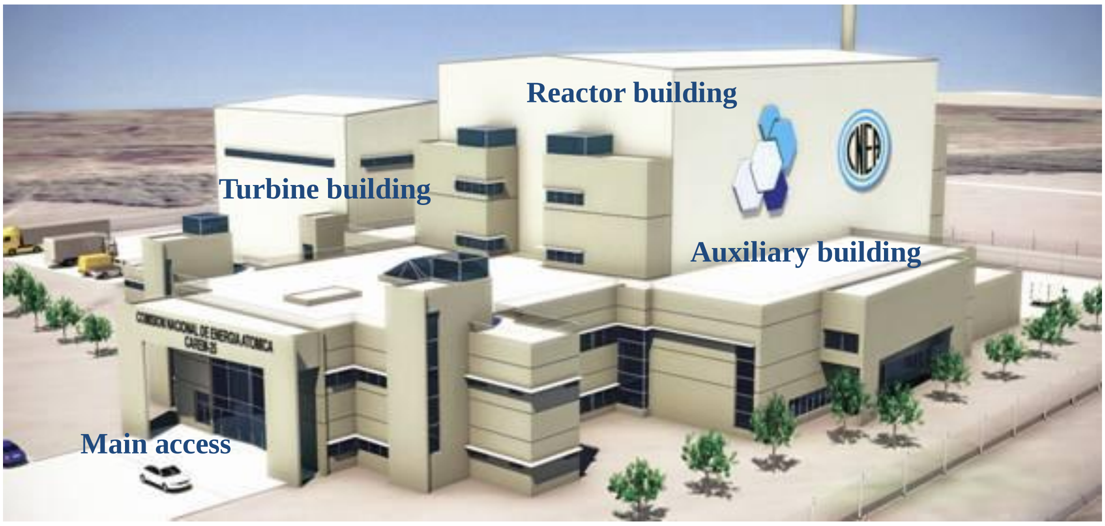
The cylindrical containment vessel with a pressure suppression pool is a 1.2 m thick reinforced concrete external wall having a stainless steel liner inner surface and withstands earthquakes of 0.25 g. Ultimate heat sink inside the containment during the grace period provides protection for external events.



Reactor pressure vessel with a safe shutdown system and residual heat removal system (Courtesy of CNEA, with permission)

Fuel Characteristics and Fuel Supply Issues

Fuel assemblies with active length of 1.4 m are arranged in a hexagonal configuration with 108 fuel rods and 19 rods equipped with in-core instrumentations, burnable poison and control rods guide tubes. The enrichment for the CAREM-25 fuel is 3.1%.



CAREM-25 plant layout (Courtesy of CNEA, with permission)

Licensing and Certification Status

The licensing process for the construction of CAREM-25 prototype was approved by the Argentina Regulatory Body (ARN) in 2010. Preliminary Safety Analysis Report and the Quality Assurance Manual were submitted to ARN (Federal Authority) for review. In September 2013, the ARN delivered the authorization to start the construction of the Prototype: Stage1, auxiliary buildings. ARN authorization to start with safety class buildings and containment constructions are expected in the 3rd quarter of 2014 (Stage2). Contracts with different Argentinean stakeholders for manufacturing of components have already been signed.

ACP-100 (CNNC, China)

Introduction

ACP-100 is an innovative design developed by China National Nuclear Corporation (CNNC) and producing power of 100 MW(e). The ACP-100 is based on existing PWR technology adapting passive safety system which uses natural convection to cool down the reactor in case of operational transients and postulated design basis accidents. The ACP-100 is an integrated PWR in which the major components of its primary coolant circuits are installed within the reactor pressure vessel (RPV). The ACP-100 plant design will allow the deployment of one to eight modules to attain larger plant output as demands arise.

Reactor System Configuration of ACP-100
(Courtesy of CNNC, with permission)

Development Milestones

2011	CNNC signed an agreement with the Zhangzhou municipal government in Fujian Province to host the first two ACP-100 demonstration units
2012	July, CNNC signed an agreement with the Chenzhou municipal government in Hunan Province to host an ACP-100 power plant
2013	Basic design
2014	Preliminary Safety Assessment Report (PSAR) expected to be approved
2015	Detailed design for construction anticipated in 2017 in Fujian

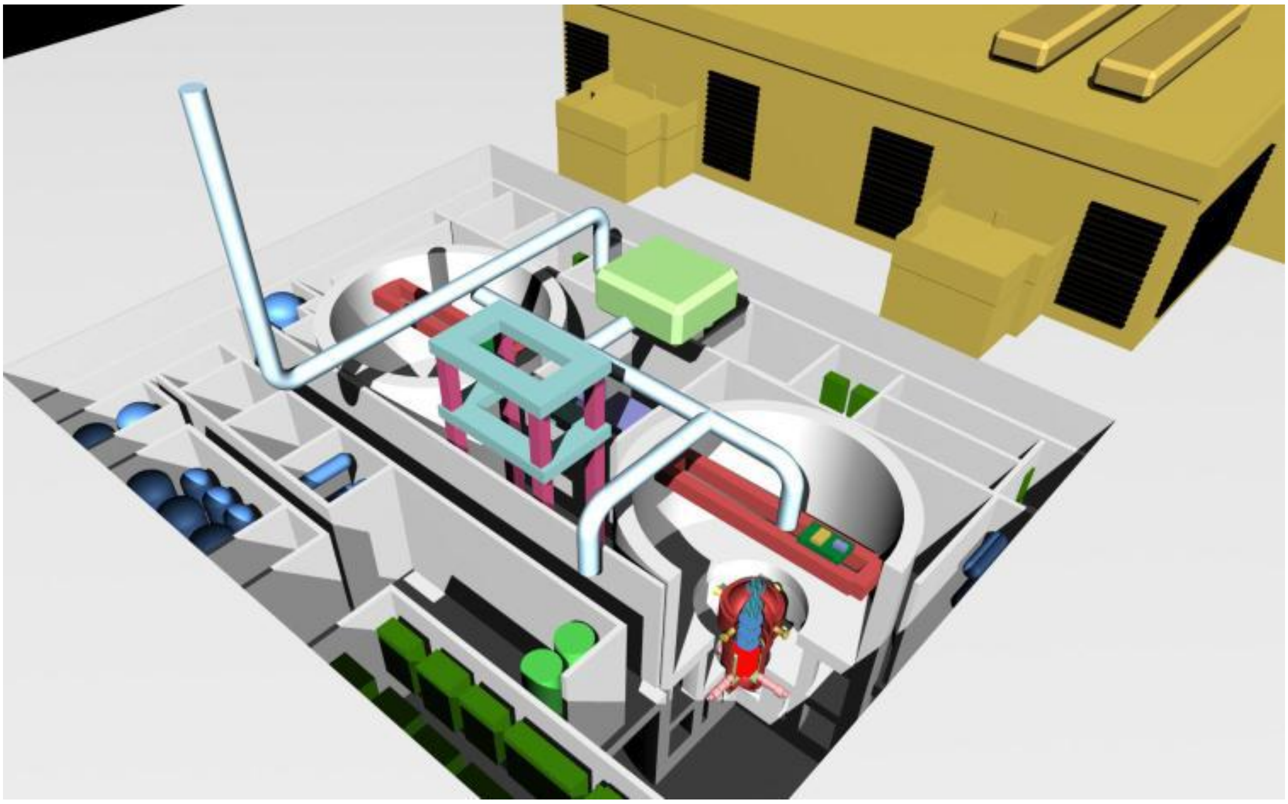
Target Applications

The ACP-100 is a multipurpose nuclear energy resource designed for electricity production, heating, steam production or seawater desalination and is suitable for remote areas that have limited energy options or industrial infrastructure.

Specific Design Features

ACP-100 concept is a modular design with vertically mounted reactor coolant pump (RCP) and control rod driving mechanism (CRDM) of magnetic force type. The pressurizer of ACP-100 is located outside of the reactor vessel. Reactor building and spent fuel pool of ACP-100 reactor design are located under the ground level to enhance better protection from external events and reduce the radioactive material release.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	CNNC (NPIC/CNPE)
Country of Origin:	China
Reactor Type:	Integral PWR
Electrical Capacity (MW(e)):	100
Thermal Capacity (MW(th)):	310
Expected Capacity Factor (%):	95
Design Life (years):	60
Plant Footprint (m^2):	200 000
Coolant/Moderator:	Light water
Primary Circulation:	Forced circulation
System Pressure (MPa):	15.0
Main Reactivity Control Mechanism:	Control rod driving mechanism (CRDM), solid burnable poison and boron
RPV Height (m):	10
RPV Inner Diameter (m):	3.19
Coolant Temperature, Core Outlet (°C):	323.4
Coolant Temperature, Core Inlet (°C):	282.6
Integral Design:	Yes
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	N/A
Low-Temp Process Heat:	Yes
Cogeneration Capability:	Possible
Design Configured for Process Heat Applications:	Heating, desalination
Passive Safety Features:	Yes
Active Safety Features:	N/A
Fuel Type/Assembly Array:	UO_2/17x17 square
Fuel Active Length (m):	2.15
Number of Fuel Assemblies:	57
Fuel Enrichment (%):	2.4 – 3.0
Fuel Burnup (GWd/ton):	<45000
Fuel Cycle (months):	24
Number of Safety Trains:	2
Emergency Safety Systems:	Passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	40
Distinguishing Features:	Integrated reactor with tube-in-tube OTSG, NI under the ground level
Modules per Plant:	1 – 8
Estimated Construction Schedule (months):	36
Seismic Design (g):	0.3
Predicted Core Damage Frequency (per reactor year):	Less than 10^{-6}
Design Status:	Detailed design



ACP-100 Reactor Building underground location (Courtesy of CNNC, with permission)

Safety Features

The integrated ACP-100 is the third generation PWR, having inherent safety characteristics of eliminating large bore primary coolant piping so large break loss of coolant accidents (LOCA) are also eliminated. ACP-100 passive decay heat removal system provides cooling for 3 days without operator intervention or 14 days with water supply from the cooling pool, drained by gravity force.
Enhanced safety and physical security of ACP-100 are made possible by installing the reactor building and spent fuel pool below ground level. The reactor vessel and equipment layout are designed to enable the natural circulation between reactor core and steam generator. The ACP-100 has a safety related direct current (DC) power source to support accident mitigation for up to 72 hours, along with auxiliary power units to recharge the battery system for up to seven days. When the spent fuel pool is filled with the spent fuel of ten years, the cooling system is able to cope for seven days of cooling in case of accident before boiling dry and uncovering the fuel.

Fuel Characteristics and Fuel Supply Issues

The 57 fuel assemblies (FAs) of ACP-100 core with total length of 2.15 m core have a squared 17x17 configuration. The expected average fuel enrichment is about 2.4-3.0 %. The reactor will be able to operate around 24 months per fuel cycle. In August 2012, Jianzhong nuclear fuel fabrication plant, subsidiary of CNNC at Yibin, manufactured FAs and control rod samples for ACP-100 R&D project at NPIC.

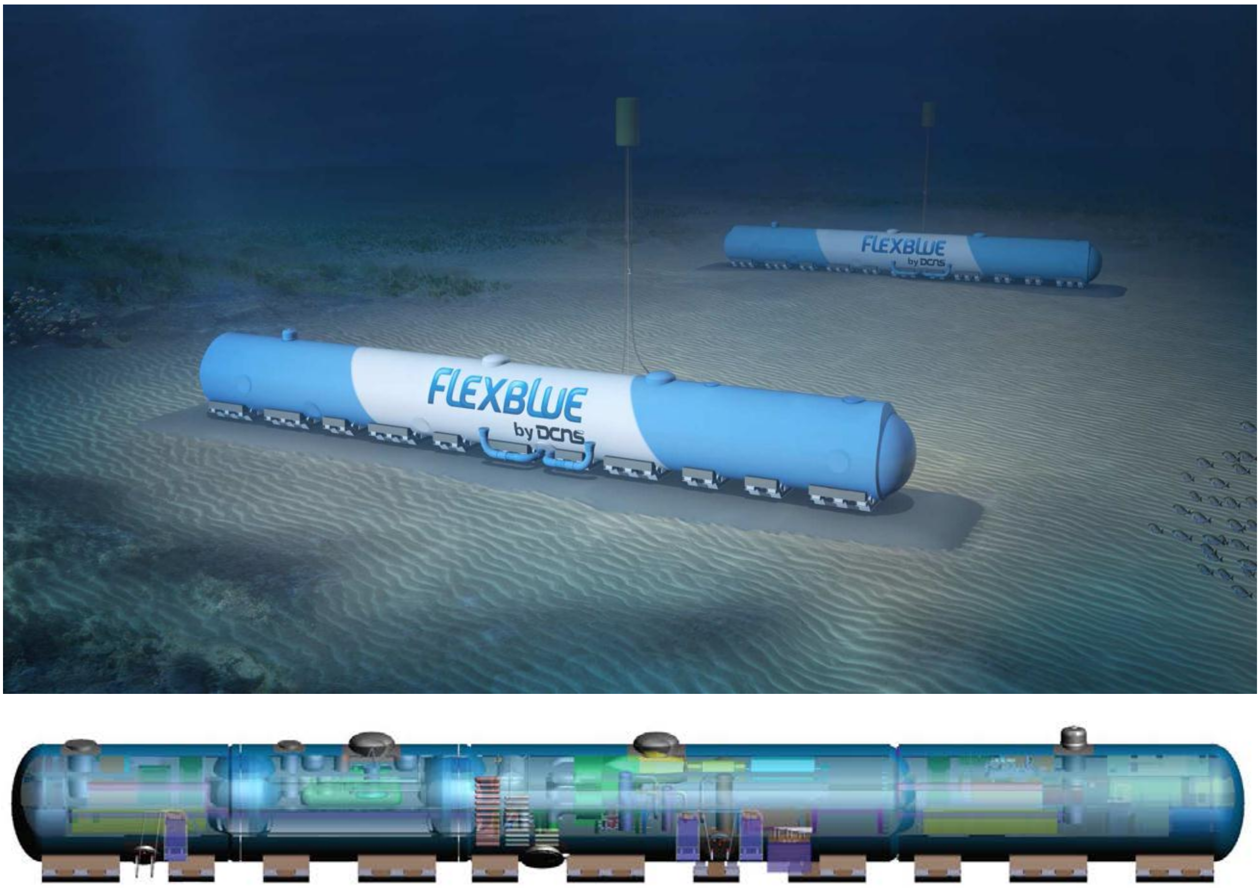
Licensing and Certification Status

The ACP-100 engineering design is close to completion, and a preliminary safety assessment report (PSAR) was recently approved. Passive emergency cooling system, control rod driving system, and critical heat flux have been tested. Control rod drive line cold tests and passive containment cooling system tests are still underway. CNNC is to submit a project proposal to the National Development and Reform Commission (NDRC) for approval. Behind that, a Nuclear Power Plant Site Safety Assessment Report and a Siting Stage Environmental Impact Assessment Report to China's regulator are supposed to be submitted for review. It is expected to start the demonstration project with the first two units for Putan, Fujian Province on the east coast of China in 2014.

Flexblue (DCNS, France)

Introduction

Flexblue is a subsea small modular power plant with an output capacity of 160 MW(e). Each module, with the length of 146 m and 14 m diameter, is moored on a stable seafloor at a depth of up to 100 m within territorial waters. Additional modules can be installed as demand increases. Flexblue is designed to be remotely operated from an onshore control room. However, each plant includes an on-board control room giving operators local control over the critical operations, including start up and maintenances. The plant is accessible by underwater vehicle when submerged. Underwater power cables transport electricity generated from Flexblue power plants to the coast and local power distribution network. Flexblue is fully manufactured in factory and shipyard and then transported by the means of the special tug or heavy lift carrier ship. Refuelling, major maintenance or dismantling also required transportation to a local support shipyard. The immersion of Flexblue power plant provides an infinite and permanently available heat sink, which, combined with passive safety systems, guarantees an infinite and natural core and containment cooling as well as protection against external hazards (airplane crashes, tsunamis, hurricanes etc.) and protection for malevolent actions.



Flexblue plant layout (Courtesy of DCNS, with permission)

Development Milestones

Target deployment by 2025.

Target Applications

Flexblue is designed to supply electricity to coastal grids.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	DCNS
Country of Origin:	France
Reactor Type:	PWR
Electrical Capacity (MW(e)):	160
Thermal Capacity (MW(th)):	530
Expected Capacity Factor (%):	90
Design Life (years):	60
Plant Footprint (m^2):	N/A for submerged modules
Coolant/Moderator:	Light water
Primary Circulation:	Forced circulation
System Pressure (MPa):	15.5
Main Reactivity Control Mechanism:	Control rods and solid burnable poison
RPV Height (m):	7.65
RPV Diameter (m):	3.84
Coolant Temperature, Core Outlet (°C):	318
Coolant Temperature, Core Inlet (°C):	288
Integral Design:	2 loops
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	N/A
Low-Temp Process Heat:	N/A
Cogeneration Capability:	N/A
Design Configured for Process Heat Applications:	N/A
Passive Safety Features:	Yes
Active Safety Features:	Non safety-grade
Fuel Type/Assembly Array:	UO_2 and Zircaloy cladding/17x17 rods in square assembly
Fuel Active Length (m):	2.15
Number of Fuel Assemblies:	77
Fuel Enrichment (%):	4.5
Fuel Cycle (months):	40
Number of Safety Trains:	2 hydraulic trains and 2 x 3 I&C channels
Emergency Safety Systems:	Passive
Residual Heat Removal Systems:	Passive and active
Refuelling & Maintenance Outage (days):	30
Distinguishing Features:	Transportable NPP, submerged operation
Modules per Plant:	Up to 6 per onshore main control room
Estimated Construction Schedule (months):	36
Seismic Design:	Not comparable – subsea conditions
Predicted Large Early Release Frequency (per reactor year):	$< 10^{-7}$
Design Status:	Conceptual design

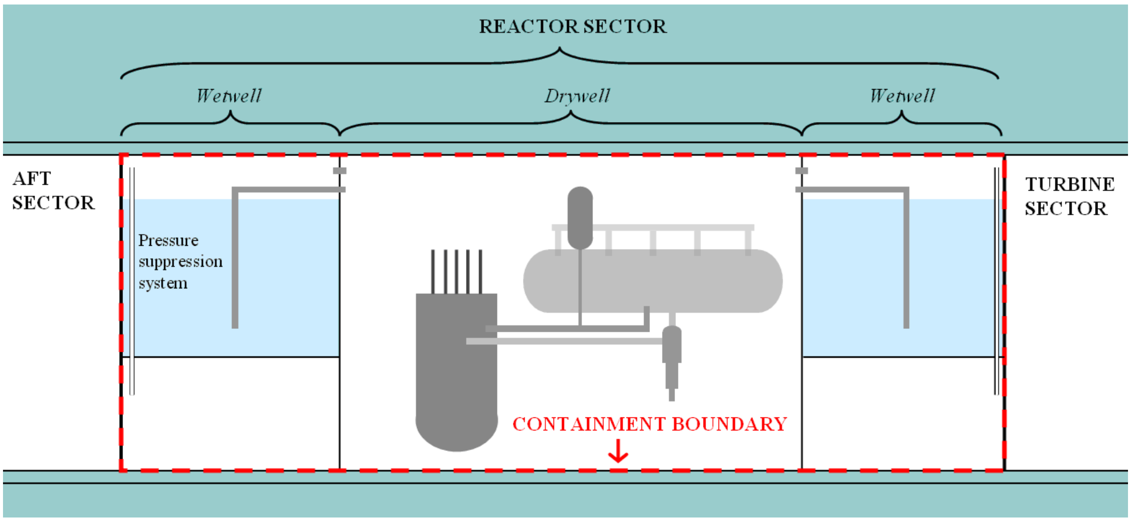
Specific Design Features

Flexblue power production cycle lasts 40 months. At the end of a production cycle, the unit is taken back to its support facility. The reactor is then refuelled and periodic maintenance is carried out. Major overhaul is scheduled every 10 years. The reactor is in shutdown condition for transport. Only the systems required for residual heat removal, control and monitoring are required in navigation conditions. 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 waste generation. A Flexblue module is composed of a turbine and alternator section, an aft section and a fore section. The two latter sections accommodate: emergency batteries, a secondary control room, process auxiliaries, I&C control panels, spares, living areas for a crew, and emergency rescue devices. Redundant main and auxiliary submarine cables transport electricity as well as information between the module and the onshore control center. Several Flexblue units can operate on the same site and hence share the same support systems.

Safety Features

DCNS claims that Flexblue satisfies international nuclear safety standards and requirements that is further 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. In particular, the nil weight/nil buoyancy of the submerged unit allows for extremely efficient de-correlation from the seabed in case of earthquakes. Furthermore, at the depth the unit is fixed, tsunami effects are not critical. Still, even in postulated extreme situations like large early release of radioactivity in the water, atmospheric release would be so reduced that it practically excludes any quick health impact on populations: water quality would have to be watched over, but no evacuation of population would be required.



The reactor sector boundary with its safety water tanks (Courtesy of DCNS, with permission)

The reactor containment is bounded by the reactor sector: hull on the sides and reactor sector walls on the front and on the back. A large share of the metal containment walls are therefore in direct contact with seawater, which provides very efficient cooling without the need for containment spray or cooling heat exchanger. Two large tanks of water – the safety tanks – act as intermediate heat sinks, pressure suppression pools and/or sources of coolant injection depending on the accident scenarios. In case of an accident, active systems designed for normal/shutdown core cooling or for controlling coolant inventory are used if alternating-current (AC) power is available. If not, passive safety systems are actuated automatically when emergency set points are reached. In all accident scenarios, a safe shutdown state is achieved and maintained for an indefinite period of time without the need for operator action. Emergency battery power is only required for the opening and closing of valves and for monitoring. The fourteen days of autonomous monitoring ability can be extended by re-charging the batteries.

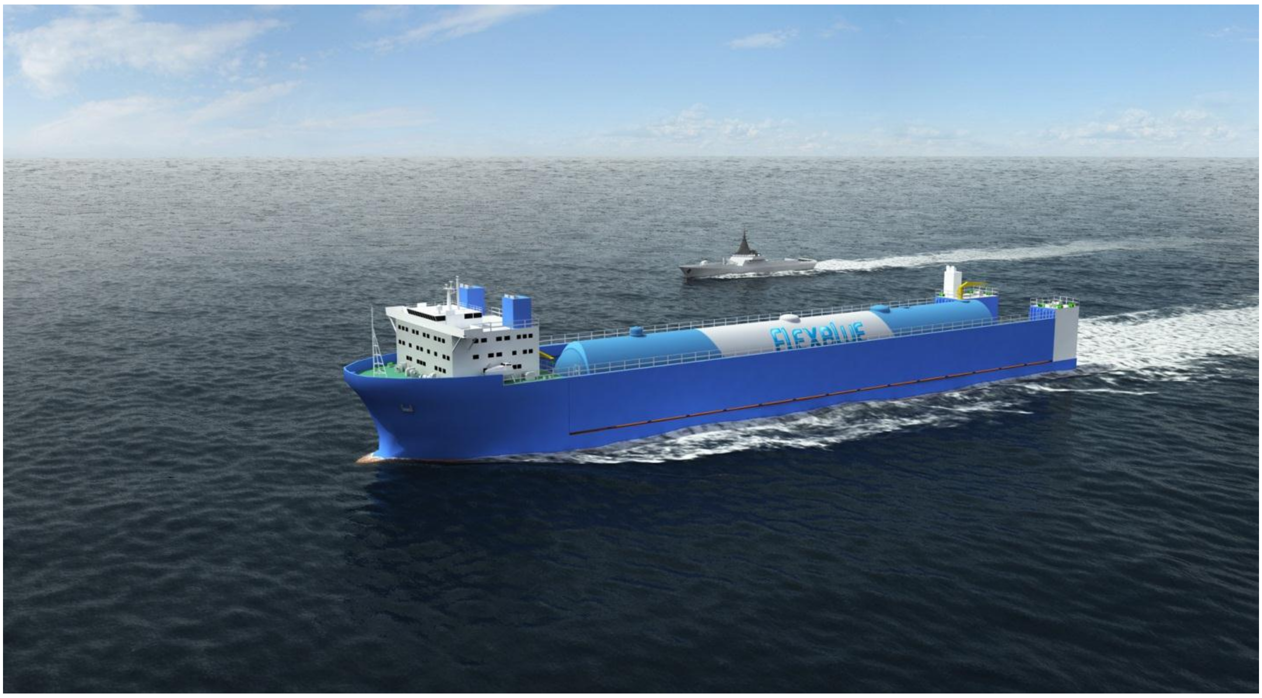
Although the safety features are designed to avoid core damage, the containment is still designed to sustain severe accident with core meltdown. In this case, the mitigation strategy consists in in-vessel corium retention assisted by passive ex-vessel core cooling.

Fuel Characteristics and Fuel Supply Issues

The reference core is made of 77 classical 17x17 fuel assemblies with an active length of 2.15m. Enrichment is kept below 5% and reactivity is controlled without soluble boron. This latter characteristic reduces the generation of radioactive wastes and simplifies the chemical control system.

Licensing and Certification Status

Presentations of the concept have been made to the French safety authority. Technical discussions have been initiated with the French technical safety authority.

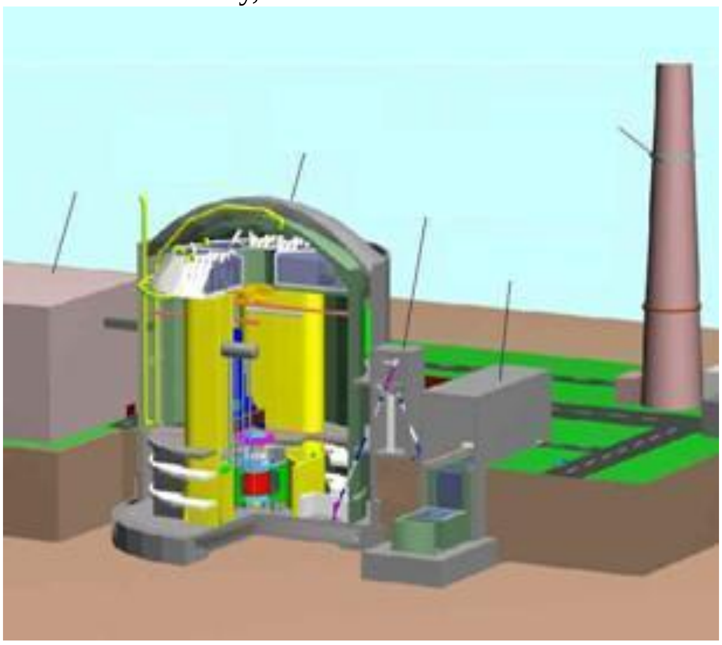


Ship transportable units (Courtesy of DCNS, with permission)

AHWR300-LEU (BARC, India)

Introduction

The Indian Advanced Heavy Water Reactor (AHWR) is a vertical pressure tube type boiling light water cooled and heavy water moderated reactor. The reactor has been designed by Bhabha Atomic Research Centre (BARC) mainly for the purpose of demonstration of large scale power production from thorium with Low Enriched Uranium (LEU) as an external feed. The safety goal of AHWR is to satisfy objectives of next generation nuclear reactor systems, such as reducing the probability of severe accidents and radioactivity release to near insignificant levels. Several innovative and passive technologies have been engineered into the reactor design such as adequate safety margins are available during normal reactor operation, decay heat removal, emergency core cooling, and confinement of radioactivity, etc.



Layout of AHWR (Courtesy of BARC, with permission)

Target Applications

The AHWR is designed for electricity production.

Specific Design Features

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 Main Heat Transport (MHT) system has been designed with the objective of ensuring cooling of reactor core by natural circulation under all operational states and during all postulated accident conditions. During normal power operation, the main heat transport system extracts and transports the heat produced in the fuel (located inside the coolant channel assemblies) to steam drums by natural circulation. In the steam drums, steam is separated from water by gravity. The separated steam is then transported to the turbine. In the steam drum, saturated water gets mixed with cold feed water, injected in the steam drum between two longitudinal baffles, and the mixed water flows to the reactor inlet header through the downcomers. In the event of non-availability of main condenser, the steam from the steam drums is sent to Isolation Condensers (ICs) which are submerged in a large pool of water (8000 m^3), known as Gravity Driven Water Pool (GDWP). The condensed steam is then returned back to respective steam drums. It is estimated that this inventory can remove decay heat for more than 110 days without need for external power supply. Emergency Core Cooling System (ECCS) and Gravity Driven Cooling System (GDCS) takes care of core cooling in the event of breach of MHT pressure boundary leading to loss of coolant from the system.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	BARC
Country of Origin:	India
Reactor Type:	Pressure tube type heavy water moderated reactor
Electrical Capacity (MW(e)):	304
Thermal Capacity (MW(th)):	920
Expected Capacity Factor (%):	90
Design Life (years):	100
Plant Footprint (m^2):	220000
Coolant/Moderator:	Light water/heavy water (D_2O)
Primary Circulation:	Natural circulation
System Pressure (MPa):	7.0
Main Reactivity Control Mechanism:	Rod insertion
Calandria Height (m):	5
Calandria Diameter (m):	6.9
Coolant Temperature, Core Outlet (°C):	285 with avg. steam exit quality of 19.7%
Coolant Temperature, Core Inlet (°C):	258.2
Power Conversion Process:	Rankine Cycle
High-Temp Process Heat:	N/A
Low-Temp Process Heat:	For production of 2650 m^3 per day desalinated water
Design Configured for Process Heat Applications:	N/A
Passive Safety Features:	Natural circulation for core heat removal under all operational states, passive core injection during LOCA, passive containment isolation during LOCA, Passive containment cooling, passive auto-depressurization, PARs for hydrogen management, Passive end shield and moderator cooling for managing SBO, Passive auto union of V1 and V2 volumes for ensuring core submergence, Passive poison injection, Passive corium cooling by core catcher
Active Safety Features:	Shut Down System #1, Shut Down System#2, Long Term Recirculation of spilled inventory, Moderator Liquid Poison Addition System, Containment Isolation, Hard vent system, Primary containment filtration and pump back system, secondary containment ventilation and purge recirculation system
Fuel Type/Assembly Array:	(Th-LEU) MOX with LEU of 19.75% ^{235}U enrichment/54 pins (12, 18, 24)
Fuel Active Length (m):	3.5
Number of Fuel Assemblies:	444
Fuel Enrichment (%):	4.3 (avg. ^{235}U) in each assembly
Fuel Burnup (GWd/ton):	60
Fuel Cycle:	LEU - Thorium
Number of Safety Trains:	4
Emergency Safety Systems:	Passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	On-power
Distinguishing Features:	LEU-Th fuel cycle; natural circulation core cooling
Modules per Plant:	1
Estimated Construction Schedule (months):	~ 60
Seismic Design (g):	0.2
Predicted Core Damage Frequency (per reactor year):	~ 10^{-8}
Design Status:	Basic design

Safety Features

The main purpose of AHWR development is demonstration of large scale power production from Thorium. The safety goal of AHWR is to satisfy objectives of next generation nuclear reactor systems, such as reducing the probability of severe accidents and radioactivity release to near insignificant levels. To achieve the above safety goals, several passive and inherent safety features have been introduced into the design of the reactor. The reactor fuel design envisages a negative fuel temperature coefficient of reactivity, negative power coefficient under all plant conditions and negative coolant void reactivity coefficient, which ensure that the reactor power reduces automatically in the event of any breach in coolant boundary leading to large voidage in the core or reduction in coolant inventory. The main heat transport system submerges the core, having more than 400 m^3 of water. This leads to very low power density of about 4 kW/l ensuring higher safety margins during any transients or accidents. Apart from the above, many passive safety features have been engineered into the design of AHWR such as:

- core heat removal by natural convection of the coolant during normal operation and in accidents without need of pumps;
- passively removing of decay heat of reactor by isolation condensers immersed in a large pool of water in a gravity driven water pool (GDWP) without requiring any external source of power to open valves or pumps;
- direct injection of emergency core cooling water into the fuel cluster to cool the hot fuel pins in a passive mode during loss of coolant accidents;
- passively cooling the containment for several days without needing containment blowers, fans, etc.;
- passively shutting down the reactor by the injection of poison to the moderator using a high-pressure steam in the case of a low probability event of failure of the wired shut down systems due to any malevolent actions;
- passive concrete cooling system to protect the concrete structure in a high temperature zone, etc.;
- passive cooling of moderator and end shield for several days in case of prolonged station blackout.

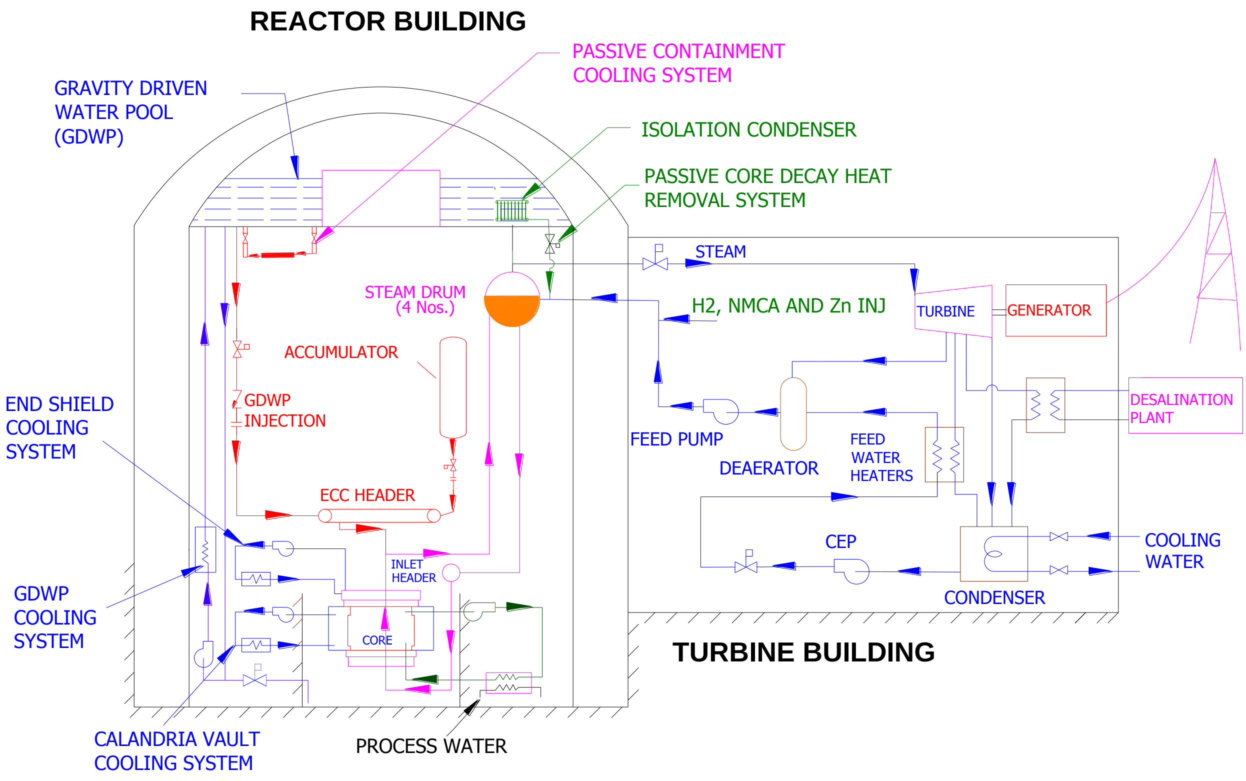
On site storage of 8000 m^3 of water located inside the containment building enables the reactor to withstand extreme events like Fukushima for more than 110 days without requiring any operator assistance.

Electrical, and Instrumentation and Control Systems

The Instrumentation and Control (I&C) of AHWR uses modern electronic technology and all the key real time control and monitoring systems are designed around computer based platforms. AHWR I&C systems include various equipment intended to perform display, monitoring, control, protection and safety functions. The general concepts followed are redundancy, diversity, physical separation of channels, fail safe behaviour, fault tolerance, testability, and maintainability. Micro controller, computer based triple modular redundant systems and triplicated hardwired systems are implemented with online testing facility. The computer based systems are connected via redundant communication network. The dual redundant server computers collect the process data from all the systems. Dedicated, distributed data acquisition and analysis systems with display workstations provide display and logging of data. Comprehensive plant information is displayed on large video screens in Main Control Room (MCR) and Back up Control Room (BCR) for the operators through dedicated servers. Uninterruptible power supplies are provided for all I&C systems. In AHWR, operator interface include large video display panels, operator work stations, alarm annunciation windows and dedicated hard wired safety panels. The design objectives met by the philosophy of “defence-in-depth” has resulted in I&C systems of high reliability and availability and they meet stringent safety and operational requirements.

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 electrical power production. 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, turbine-generator and auxiliaries, condensing system, condensate and feedwater heating system, steam dumping and relief systems and on-line full condensate flow purification.



General AHWR layout (Courtesy of BARC, with permission)

Fuel Characteristics and Fuel Supply Issues

The AHWR300-LEU circular fuel cluster consists of 54 (Th, LEU) O_2 fuel pins, 12 fuel pins in the inner ring with 30% LEU, 18 fuel pins in the middle ring with 24% LEU and 24 fuel pins in the outer ring have an average of 16% LEU. 444 fuel clusters comprise the full core.

Licensing and Certification Status

AHWR pre-licensing review has been completed by the national regulatory body which has approved AHWR design in principle. The regulatory body has recommended for experimental demonstration of the First-Of-A-Kind features. BARC has built several integral and separate effect test facilities for design validation of AHWR. In view of the post Fukushima scenario, new design features have been incorporated to meet the challenges of extended grace period of at least seven days and tolerating a prolonged station blackout with extreme events as happened in Fukushima. These include passive end shield and passive moderator cooling system, hard vent system, passive autocatalytic recombiners for hydrogen management and a core catcher. The new design features are being validated in simulated test facilities. Site selection of AHWR has been completed and the necessary clearance from competent authorities is underway. PSAR is ready for submission to regulatory body after obtaining site clearances from competent authorities.

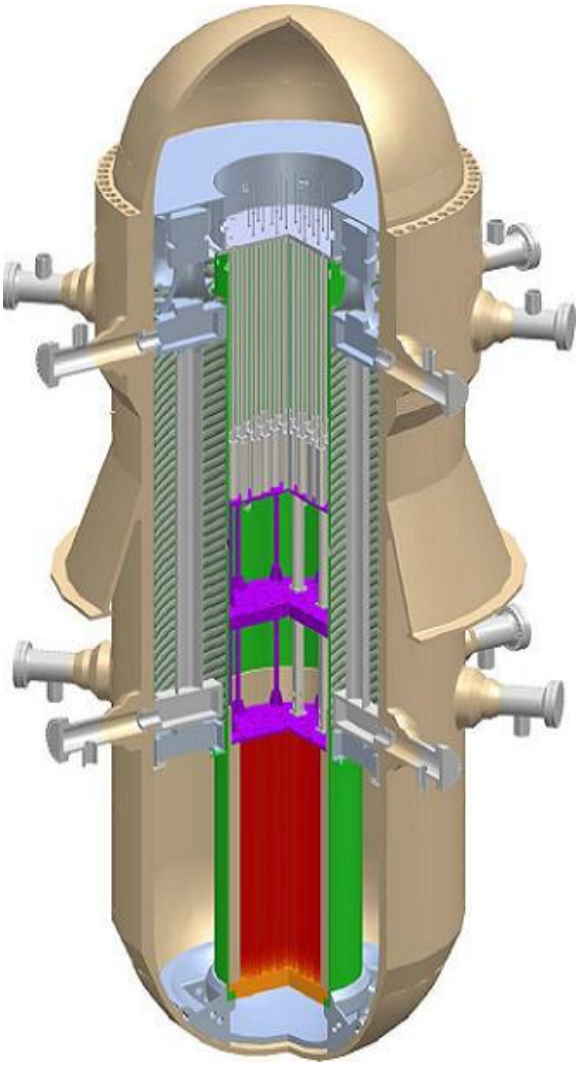
IRIS (IRIS, International Consortium)

Introduction

IRIS is an LWR with a modular, integral primary system configuration. The concept was originally pursued by an international group of organizations led by Westinghouse. Currently the IRIS related activities, especially those devoted to large scale integral testing, are being pursued by Italian organisations.

IRIS is designed to satisfy enhanced safety, improved economics, proliferation resistance and waste minimization. Its principle characteristics 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 of active and passive safety systems;
- optimized maintenance with intervals of at least four years.



Reactor System Configuration of IRIS (Courtesy of POLIMI, with permission)

Development Milestones

2001	Conceptual design completion
2001	Preliminary design start-up
2002	Pre-licensing process activities

Target Applications

The primary application of IRIS design is electricity production. However, this integrated PWR can support the heat producing process and seawater desalination options. Coupling with renewable energy parks and energy storage systems has been addressed as well.

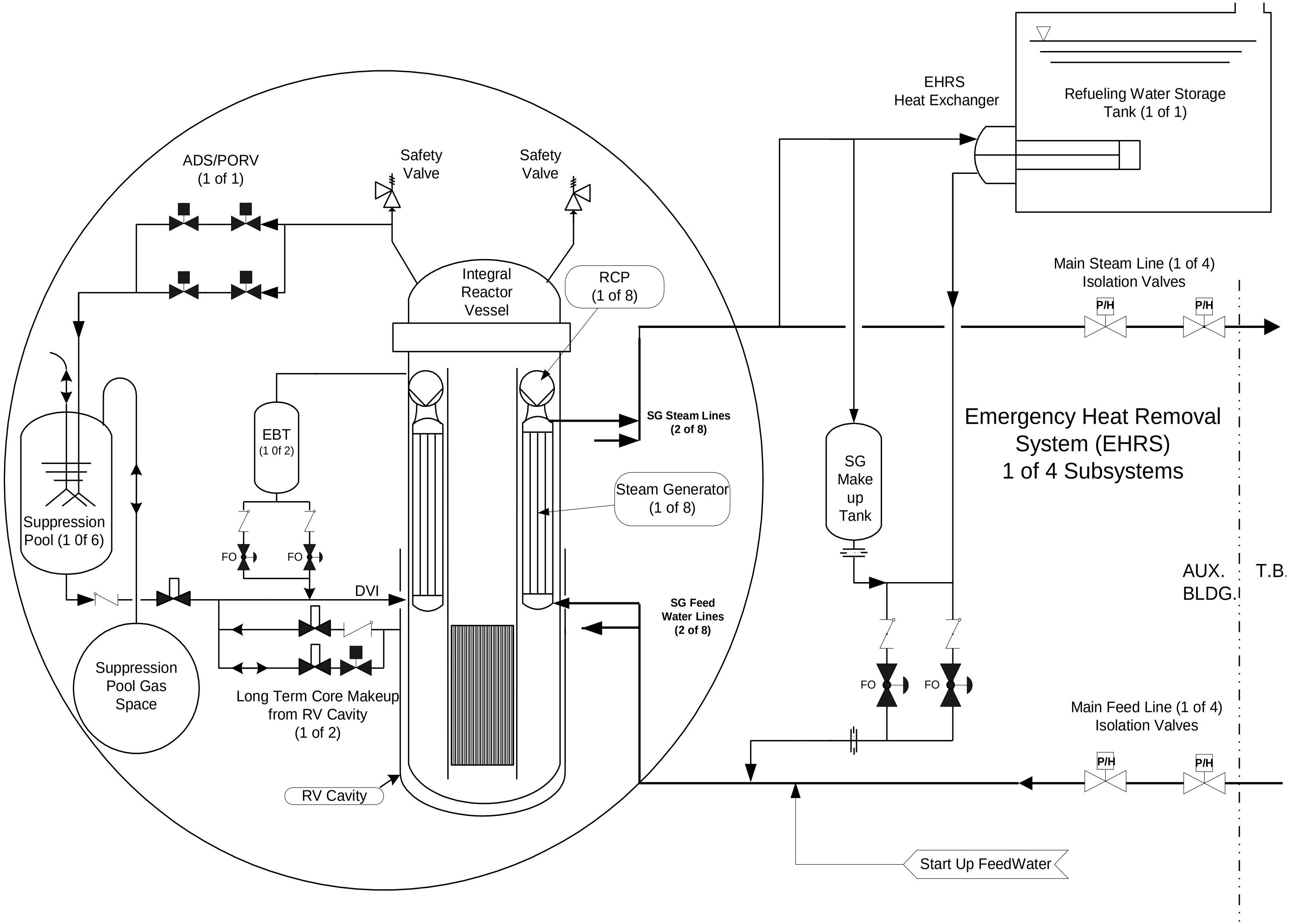
Specific Design Features

The integral reactor coolant system of IRIS consists of 8 helical-coil steam generators, 8 axial flow fully immersed primary coolant pumps, internal control rod drive mechanism (CRDM) and integral pressurizer installed within the reactor pressure vessel (RPV).

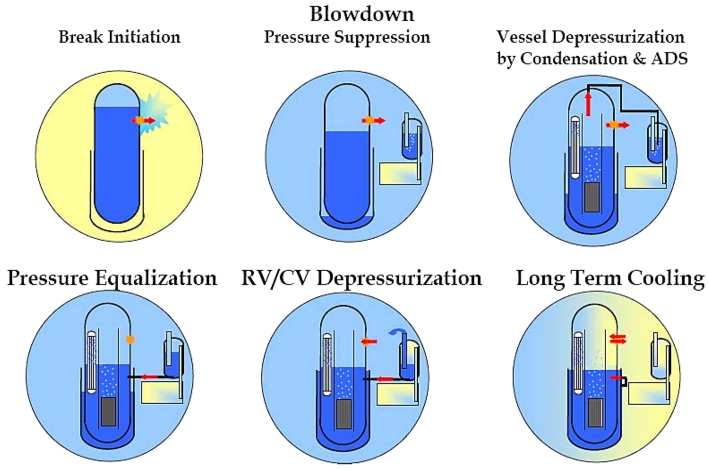
Safety Features

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 or downgraded. This provides a defence in depth that allow IRIS eliminates emergency response zone. The auxiliary building is fully seismically isolated.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	IRIS
Country of Origin:	International Consortium
Reactor Type:	Integral PWR
Electrical Capacity (MW(e)):	335
Thermal Capacity (MW(th)):	1000
Expected Capacity Factor (%):	> 96
Design Life (years):	60
Plant Footprint (m^2):	141 000 (four units layout)
Coolant/Moderator:	Light water
Primary Circulation:	Forced circulation
System Pressure (MPa):	15.5
Main Reactivity Control Mechanism:	ICRDM (Internally driven Control Rods)
RPV Height (m):	21.3
RPV Diameter (m):	6.2
Coolant Temperature, Core Outlet (°C):	330
Coolant Temperature, Core Inlet (°C):	292
Integral Design:	Yes
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	(typical for district heating/cogeneration applications)
Low-Temp Process Heat:	(typical for district heating/cogeneration applications)
Cogeneration Capability:	Yes
Design Configured for Process Heat Applications:	Yes
Passive Safety Features:	Yes
Active Safety Features:	Non-safety grade active systems
Fuel Type/Assembly Array:	UO_2/MOX/17x17 square
Fuel Active Length (m):	4.26
Number of Fuel Assemblies:	89
Fuel Enrichment (%):	4.95
Fuel Burnup (GWd/ton):	65 (max)
Fuel Cycle (months):	48 (max)
Reactivity Control:	Soluble boron and rod insertion
Number of Safety Trains:	4
Emergency Safety Systems:	Passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	N/A
Distinguishing Features:	Integral primary system configuration
Modules per Plant:	1 to 4
Estimated Construction Schedule (months):	36
Seismic Design (g):	0.3
Predicted Core Damage Frequency (per reactor year):	10^{-8}
Design Status:	Basic design

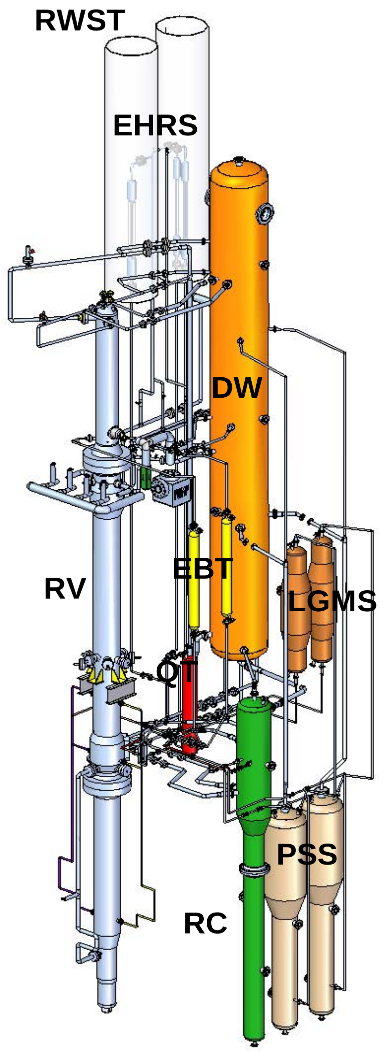


Engineered safety features of IRIS (Courtesy of POLIMI, with permission)



SBLOCA safety strategy (Courtesy of POLIMI, with permission)

This provides a high level of defence in depth that may allow IRIS to claim no need for an emergency response zone. As for auxiliary building of IRIS is that they are fully seismically isolated. The IRIS pressure suppression containment vessel has a spherical configuration and it is 25 m in diameter. In case of small break loss of coolant accident (SB LOCA), the RPV and containment become thermodynamically coupled. The pressure differential across the break equalizes quickly and LOCA is stopped. The core remains covered for all postulated breaks during the whole transient. The heat sink is designed to provide cooling for 7 days without operator action or off-site assistance for replenishing.



Integral Testing Facility under construction at SIET labs (Italy)

The once-through steam generators (OTSGs), steam and feed lines and the emergency heat removal system (EHRS) are designed for full primary pressure of 15.5 MPa. The EHRS does not inject water, but only removes heat from the reactor via the SGs. The Emergency Boration Tank (EBT) injects a limited amount of water. EBT borates primary system to maintain reactor subcritical at low temperatures, and provides diverse means of shutdown for ATWS events, as well as limited amount of water makeup following LOCA and cooldown event.

Fuel Characteristics and Fuel Supply Issues

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 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.

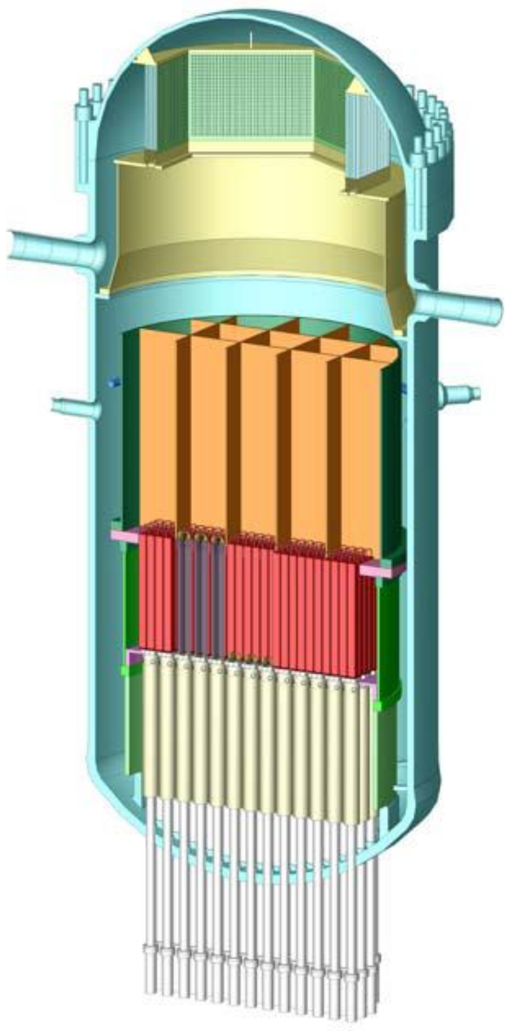
Licensing and Certification Status

The IRIS team has completed the design of the large scale test facility, currently under construction, to prepare for future design certification. R&D activities in the field of design economics, financial risk and SMR competitiveness are under way.

DMS (Hitachi-GE Nuclear Energy, Japan)

Introduction

DMS stands for Double MS: Modular Simplified and Medium Small Reactor. The concept design of this reactor has been developed by Hitachi-GE Nuclear Energy under the sponsorship of the Japan Atomic Power Company (JAPC) from 2000 to 2004. The design is small-sized boiling water reactor (BWR) which generates thermal power of 840 MW(th) or 300 MW(e). The heat from the core is removed by natural circulation so recirculation pumps and their driving power sources are eliminated. This feature allows for a simplified and compact reactor pressure vessel (RPV) and containment. Due to the natural circulation feature, reactor internals and systems are also simplified. As a defence-in-depth measure, enhanced hybrid safety systems that combine passive and active methods are adopted. Like in other BWR, steam separation is performed inside the RPV. In DMS however, this mechanism is done through free surface separation (FSS) in which the steam is separated from water by gravity force. Hence, no physical separator assembly is required.



Reactor System Configuration of DMS (Courtesy of Hitachi-GE Nuclear Energy)

Development Milestones

2000-04	Conceptual design
2014	Basic design (pre-licensing)
2024	Deployment

Target Applications

A small-to-medium sized boiling water reactor is suitable for where budget for construction is limited and electricity transmission networks have not been fully constructed. DMS design also provides a nonelectric use of energy such as for district heating, mining (oil sand extraction/SAGD) and desalination.

Specific Design Features

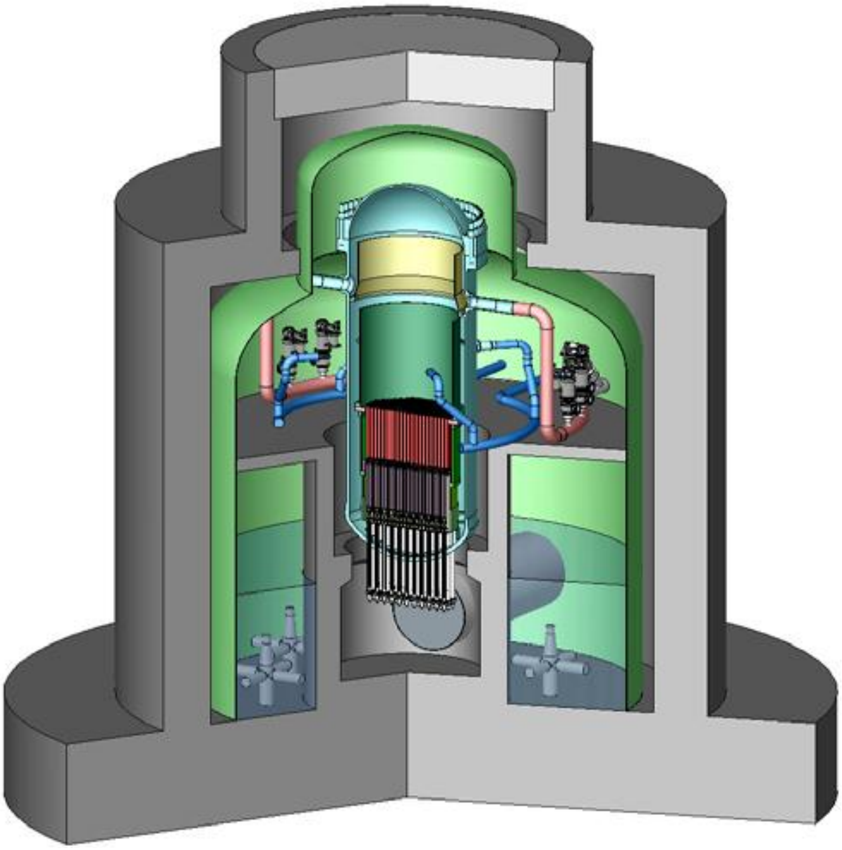
The main features of the DMS reactor design are the miniaturization and simplification of systems and equipment, integrated modulation of construction, standardization of equipment layouts and effective use of proven technology. The decrease in primary containment vessel (PCV) height is achieved by reducing the active fuel length of the DMS core, which is about two meters compared with 3.7 meters in the conventional BWR. The short active fuel length reduces the pressure drop in core and enables natural circulation. Low power density (44 kW/l) results in a moderate evaporation rate and lower steam velocity in the upper plenum of the reactor pressure vessel (RPV). This let the design to adopt a free surface separation (FSS) system. The FSS eliminates the need for a separator and thus helps minimize the RPV and PCV sizes. The designer of the plant also has anticipated multipurpose use of the produced energy by establishing advanced BOP system which allows the heat for non-electrical applications such as process heat, mining (oil sand extraction) and desalination. In addition, factory-made module is made to reduce the construction period and to be land transported.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	Hitachi-GE Nuclear Energy
Country of Origin:	Japan
Reactor Type:	Boiling Water Reactor
Electrical Capacity (MW(e)):	300
Thermal Capacity (MW(th)):	840
Expected Capacity Factor (%):	> 87
Design Life (years):	60
Plant Footprint (m^2):	N/A
Coolant/Moderator:	Light water
Primary Circulation:	Natural circulation
System Pressure (MPa):	7.171
Main Reactivity Control Mechanism:	Control rod drive and boric acid injection
RPV Height (m):	15.5
RPV Diameter (m):	5.8
Coolant Temperature, Core Outlet (°C):	287 (steam outlet)
Coolant Temperature, Core Inlet (°C):	189 (feed water)
Integral Design:	Yes
Power Conversion Process:	Direct Rankine Cycle
High-Temp Process Heat:	N/A
Low-Temp Process Heat:	Yes
Cogeneration Capability:	Yes
Design Configured for Process Heat Applications:	Yes
Passive Safety Features:	Yes
Active Safety Features:	Yes
Fuel Type/Assembly Array:	N/A
Fuel Active Length (m):	2
Number of Fuel Assemblies:	400 (short length)
Fuel Enrichment (%):	4.3
Fuel Burnup (GWd/ton):	45
Fuel Cycle (months):	24
Number of Safety Trains:	3 active systems, 1 passive system
Emergency Safety Systems:	ADS, Hybrid RCIC, LPFL, RHR, IC, PCCS
Residual Heat Removal Systems:	2 Trains RHR
Refuelling Outage (days):	N/A
Distinguishing Features:	Simple reactor design, Natural circulation system, Hybrid safety system, Multipurpose use of energy
Modules per Plant:	1
Estimated Construction Schedule (months):	~24
Seismic Design:	N/A
Predicted core damage frequency (per reactor year):	10^{-8}
Design Status:	Basic design

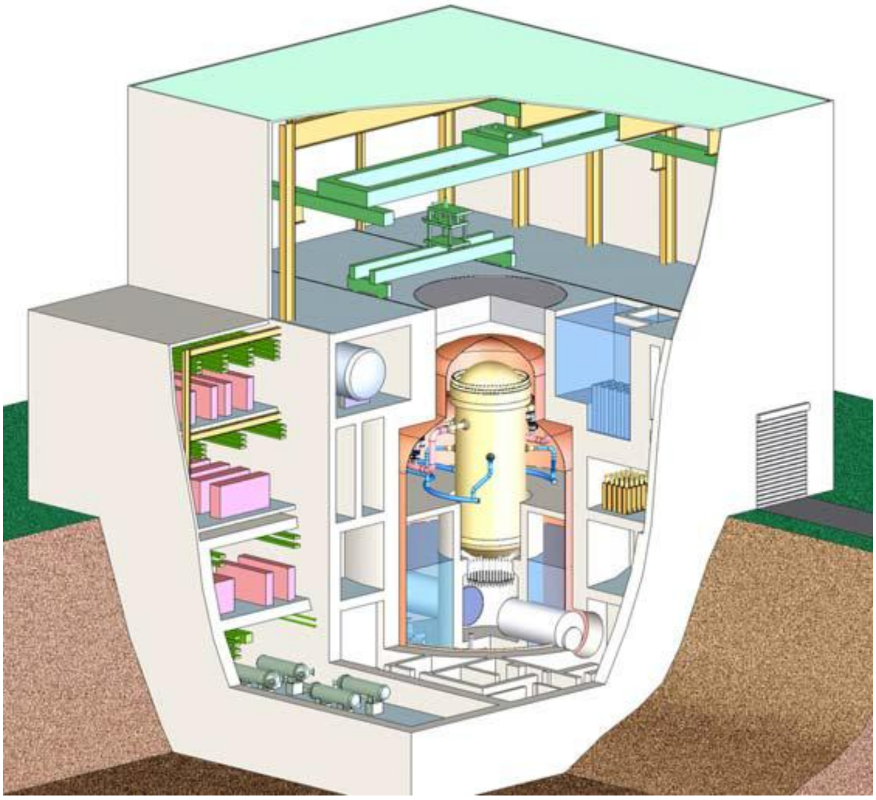
Safety Features

The basis of Emergency Core Cooling System (ECCS) design for the DMS is to attain improvement in safety efficiency. The ECCS configuration is examined to optimize core coverage capability and economic efficiency by eliminating high-pressure injection systems, adopting passive safety-related systems and adjusting distribution for the systems and power source for the ECCS. In this way the ECCS for the DMS is expected to have the same level of safety as in the ABWR while at the same time also has economic gain. Based on the results of Loss of Coolant Accident (LOCA) analysis, core coverage can be achieved by this configuration. Therefore, the plant concept was found to offer both economic efficiency and safety.

The safety system configuration of DMS has been rationally simplified compared to a conventional large BWR (ABWR). There are 4 main distinctive features of the system: (1) High pressure core flooder (HPCF) equipped in conventional ABWR is eliminated because DMS has larger coolant inventory in reactor pressure vessel (RPV) than ABWR; (2) Isolation condenser (IC) and passive containment cooling system (PCCS) are added to the active system as a countermeasure against long-term SBO. IC and PCCS can passively remove the decay heat during at least 10 days; (3) Gas turbine generator (GTG) was adopted instead of conventional diesel generator (D/G). GTG includes less auxiliary equipment than D/G, so maintenance load decreases and reliability increases. In case of ABWR, GTG cannot be applied because of slow start-up characteristic. On the other hand, DMS can adopt GTG because DMS has large time margin until water level in the RPV reaches below the top of an active fuel; and (4) Reactor core isolation cooling (RCIC) system and low pressure core flooder (LPFL) system were rationally integrated as Hybrid RCIC. RCIC can inject water into the RPV by using steam generated in the RPV under high RPV pressure and LPFL can inject water by motor-driven pump under low RPV pressure. The hybrid RCIC can inject water by using steam power under high RPV pressure, and by using electric power under low RPV pressure. Long-term SBO and design basis accident (DBA) were preliminary analyzed and it was confirmed that the core could be cooled for 10 days against SBO and peak cladding temperature (PCT) was kept less than 1200℃ even against the most severe DBA.



Compact primary containment vessel of DMS (Courtesy of Hitachi-GE Nuclear Energy)



Standardized Reactor Building (Courtesy of Hitachi GE Nuclear Energy)

Building Layout

The reactor building is minimized by both system simplification and PCV compactness. The number of system component is reduced by adoption of large capacity equipment, common use of single equipment for different system, and adoption of passive system. PCV compactness is achieved by dish shape drywell and eccentric RPV arrangement, i.e., the RPV is installed not at the real center but at an eccentric center of the PVC. Compact PCV lets the number of floor levels to reduce from six in current ABWR's to four, which contribute to saving in the construction period. The building is divided into fixed standard area, where hardly influenced by site conditions and variable flexible area which may depend on site conditions. The main power block surrounding the PCV or the secondary containment is designed to be the standard area. On the other hand, the circumferential area such as the electrical room, plant make up facilities, etc. is designed to be flexible area. By this approach of rationalized layout, it is possible to realize the rate of building volume per unit output power that is equivalent to ABWR.

Fuel Characteristics and Fuel Supply Issues

The 840 MW(th) reactor core is loaded with 400 fuel bundles. Each has 2 m active length with 4.3 wt% of enrichment. Core power density is about 44 kW/l. The core can produce energy with the refuelling period of 24 months. The fission reaction in the core is controlled by 137 control rods inserted from below the vessel.

Licensing and Certification Status

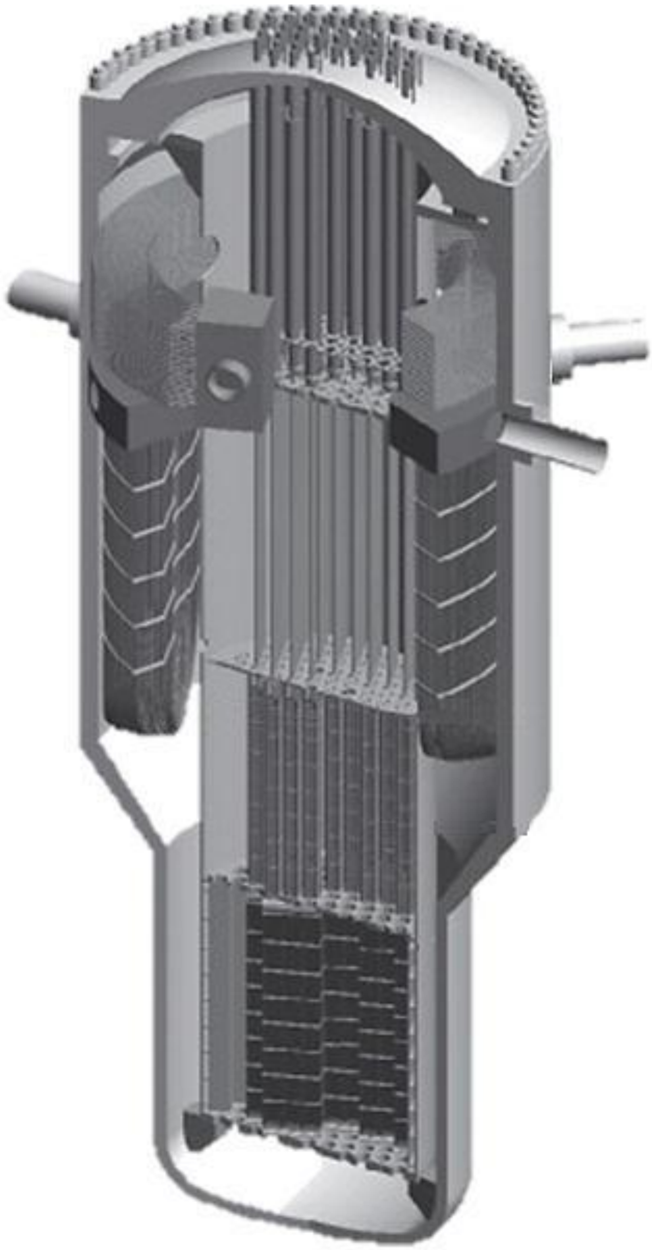
At the moment, no domestic license or pre-license activities for SMR in general, since there is no SMR construction project in Japan. Some SMR design applied or will apply for a pre-licensing in USA or Canada (i.e. in Customer's country).

(Note: Japan does not have a process of 'vendor's design assessment', such as design certification in the United States or generic design assessment in the United Kingdom. Vendor's design is reviewed in the license process of the Establishment Permit (Construction license) of the NPP construction project).

IMR (Mitsubishi Heavy Industries, Japan)

Introduction

The Integrated Modular Water Reactor (IMR) is a medium sized power reactor with a reference output of 1000 MW(th) and producing electricity of 350 MW(e). This integral primary system reactor (IPSR) set a potential deployment after 2025. IMR employs the hybrid heat transport system (HHTS), which is a natural circulation system under bubbly flow condition for primary heat transportation, and no penetrations in the primary cooling system by adopting the in-vessel control rod drive mechanism (CRDM). These design features allow the elimination of the emergency core cooling system (ECCS).



Reactor System Configuration of IMR
(Courtesy of MHI, with permission)

Development Milestones

1999	IMR started its conceptual design study at Mitsubishi Heavy Industries (MHI).
2001-04	An industry-university group led by MHI, including Kyoto University, Central Research Institute of Electric Power Industries (CRIEPI), the Japan Atomic Power Company (JAPC), and MHI were developing related key technologies through two projects, funded by the Japan Ministry of Economy, Trade and Industry. In the first project, the feasibility of the HHTS concept was tested through two series of experiments.
2005-07	In the second project, the thermal-hydraulic data under natural circulation conditions with bubbly flow for the HHTS design were obtained by four series of simulation tests using alternate fluids, sulfur hexafluoride (SF_6) gas and ethanol (C_2H_5OH), whose pairing has good agreements with steam-water under high pressure conditions around 8~15MPa on bubbly flow performances.

Target Applications

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

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	Mitsubishi Heavy Industries
Country of Origin:	Japan
Reactor Type:	Integral PWR
Electrical Capacity (MW(e)):	350
Thermal Capacity (MW(th)):	1000
Expected Capacity Factor (%):	> 87
Design Life (years):	60
Plant Footprint (m^2):	N/A
Coolant/Moderator:	Light water
Primary Circulation:	Natural circulation
System Pressure (MPa):	15.51
Main Reactivity Control Mechanism:	Rod insertion
RPV Height (m):	17
RPV Diameter (m):	6
Coolant Temperature, Core Outlet (°C):	345
Coolant Temperature, Core Inlet (°C):	329
Integral Design:	Yes
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	N/A
Low-Temp Process Heat:	Possible
Cogeneration Capability:	Possible
Design Configured for Process Heat Applications:	Possible
Passive Safety Features:	Yes
Active Safety Features:	Yes
Fuel Type/Assembly Array:	UO_2 pellet/21x21 square
Fuel Active Length (m):	2.4
Number of Fuel Assemblies:	97
Fuel Enrichment (%):	4.8
Fuel Burnup (GWd/ton):	> 40
Fuel Cycle (months):	26
Number of Safety Trains:	4
Emergency Safety Systems:	Passive
Residual Heat Removal Systems:	Active
Refuelling Outage (days):	< 30
Distinguishing Features:	Steam generators in liquid and vapour regions of the vessel
Modules per Plant:	1
Estimated Construction Schedule (months):	< 42
Seismic Design:	Equivalent to that of PWRs in Japan
Predicted Core Damage Frequency (per reactor year):	2.9×10^{-7}
Design Status:	Conceptual design completed

Specific Design Features

The main features of the IMR plant system are:

- No reactor coolant pumps, pressurizer and coolant pipes with large diameters,
- The small containment vessel (CV) is made possible by adopting the integrated primary system design and simplified systems,
- A simplified chemical and volume control system (CVCS) and waste disposal system (WDS) achieved by the boric-acid free design,
- No ECCS and containment cooling/spray system,
- Simple support systems, such as the component cooling water system (CCWS), the essential service water system (ESWS) and the emergency AC power system. These are designed as non-safety grade systems, possible by use of a stand-alone diesel generator.

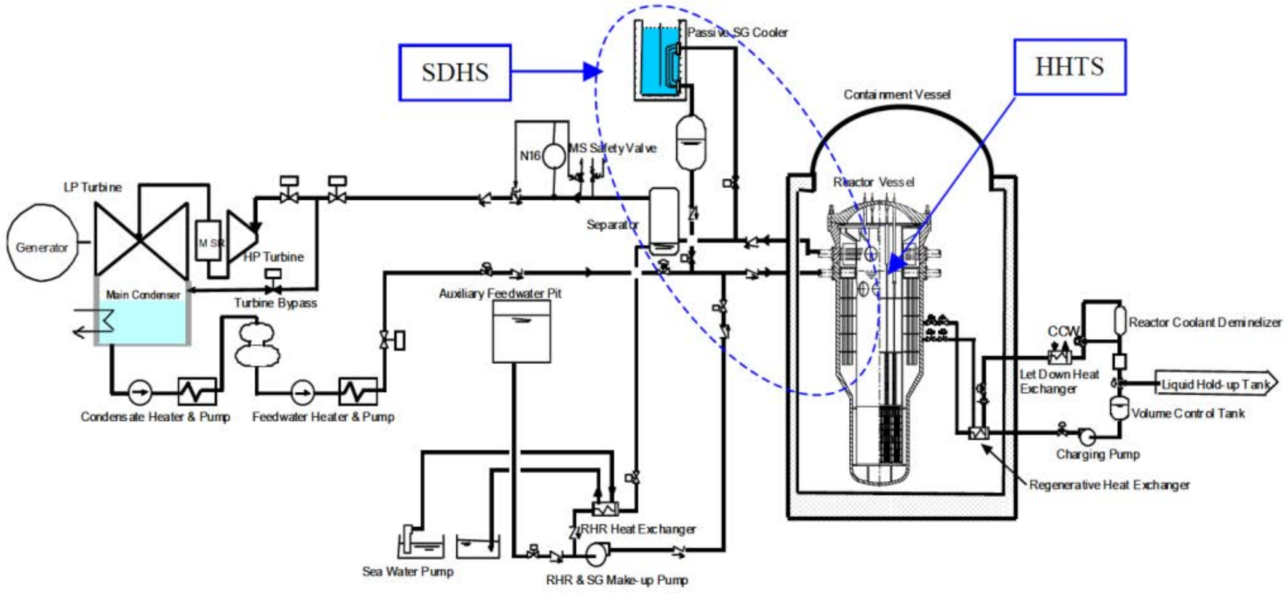
In the HHTS, IMR employs natural circulation and a self-pressurized primary coolant system, altogether resulting in a simple primary system design without reactor coolant pumps and pressurizer, it also reduces maintenance requirements. In addition, the use of the HHTS concept makes it possible to reduce the size of the RV. The HHTS is a kind of two-phase natural circulation system. The coolant starts boiling in the upper part of the core, and two-phase coolant keeps bubbly flow and flows up in the riser and is condensed and cooled by the SGs. This design approach increases coolant flow rate and thus reduces the required the RV height to transport the heat from the core. The IMR primary cooling system design under bubbly flow makes it easy to employ PWR design technologies.

Control rods whose neutron absorber is 90 wt% enriched B4C 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.

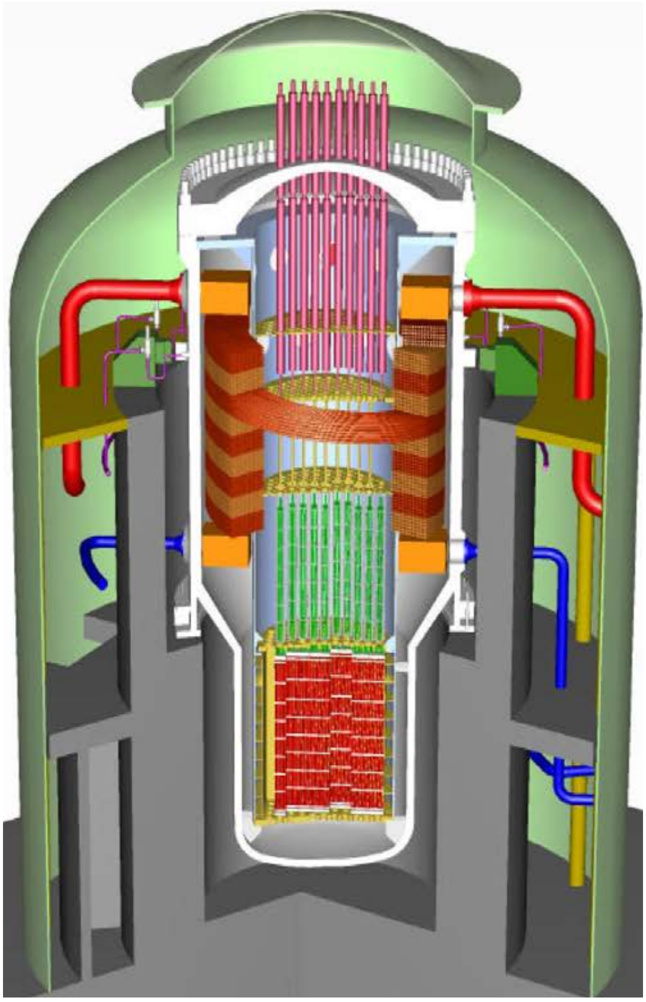
Safety Features

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.

Design basis accidents (DBAs) are supposed to cause no fuel failure with the functions of reactor shutdown and residual heat removal being carried out by the SGs cooling system. In case of beyond-design-basis accidents (BDBA) such that the SGs cooling system is not available, normal cooling systems, such as the component cooling water system (CCWS), residual heat removal system (RHRS), etc. are used if possible. In case normal cooling systems are unavailable, the RV integrity is retained by core cooling through the RV wall, and submerging the CV head, using the water for refuelling, retains the CV integrity. When decay heat removal through the SGs is not applicable, water leaking out of the RV will fall to the bottom of the RV cavity. Since decay heat can be removed through the RV wall, molten core debris could be retained inside the RV. In addition, decay heat in the CV could be removed through the CV head, which will be immersed in water supplied from the raw water tank by the operators, to keep the pressure in the CV lower than the design limitation. In the severe accidents, the RV decompression valve and the primary relief valve operate to reduce the pressure in the RV when accidents raise the primary pressure. The CV water injection line from the refuelling water pit is also used to make a heat sink in the CV at the severe accidents.



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



IMR containment layout
(Courtesy of MHI, with permission)

Fuel Characteristics and Fuel Supply Issues

The IMR has a core consisting of ninety-seven 21×21 fuel assemblies with an average enrichment of 4.95%.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.

Licensing and Certification Status

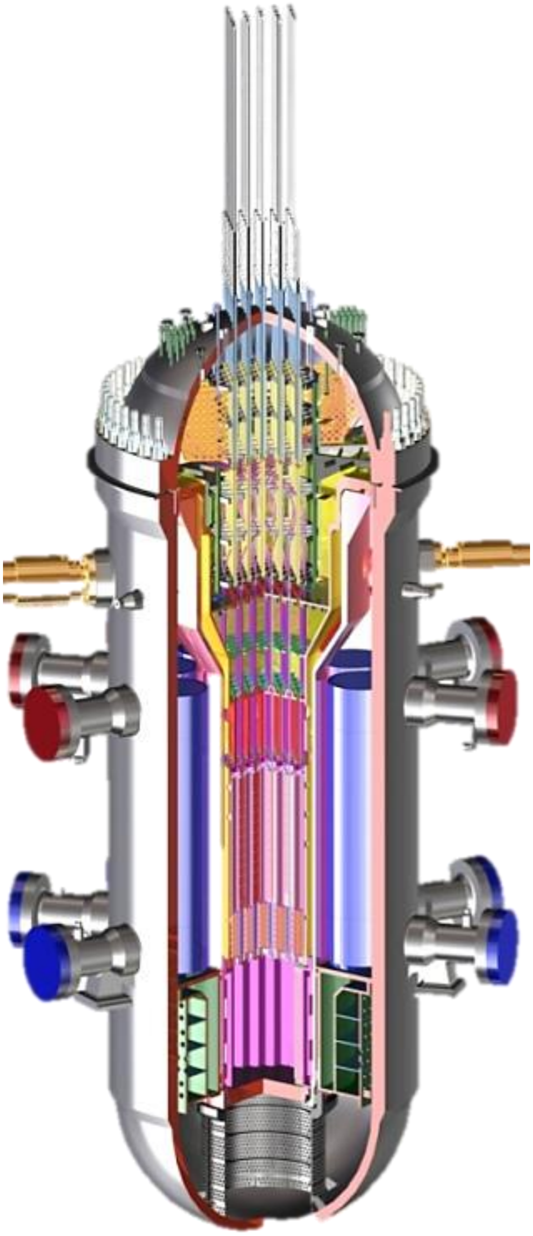
The IMR conceptual design study was initiated in 1999 by Mitsubishi Heavy Industries (MHI). A group led by MHI and including Kyoto University, the Central Research Institute of the Electric Power Industry and the Japan Atomic Power Company, developed related key technologies through two projects, funded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

SMART (KAERI, Republic of Korea)

Introduction

System Integrated Modular Advanced ReacTor (SMART) is a small integral PWR with a rated power of 330 MW(th) or 100 MW(e). To enhance safety and reliability, the design configuration has incorporated inherent safety features and passive safety systems. The design aim is to achieve improvement in the economics through system simplification, component modularization, reduction of construction time and high plant availability. By introducing a passive residual heat removal system (PHRS), presented on Figure 2, and an advanced mitigation system for loss of coolant accidents (LOCA), significant safety enhancement can be expected. 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.



Reactor System Configuration of SMART
(Courtesy of KAERI, with permission)

Development Milestones

1999	Conceptual Design Development
2002	Basic Design approval (PSA)
2007	SMART-PPS (Pre-Project Service)
2012	Technology verification, SDA (Standard Design Approval)
2012	First step of Post-Fukushima Corrections and Commercialization

Target Applications

SMART is applicable for electricity production suitable for small or isolated grids and heat district as well as processing heat for desalination purposes having the output enough to meet demands for a city of 100000 population.

Specific Design Features

SMART integrated design implies that the reactor pressure vessel contains all of the primary components such as core structures, steam generators (SGs), and the reactor coolant pumps horizontally mounted into the reactor vessel. The pressurizer functions by utilizing the volume formed in the top of the reactor pressure vessel head. The reactor coolant pumps actively recirculate the coolant routing it into the core after having transferred heat to the secondary side of the SGs.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	KAERI
Country of Origin:	Republic of Korea
Reactor Type:	Integral PWR
Electrical Capacity (MW(e)):	100
Thermal Capacity (MW(th)):	330
Expected Capacity Factor (%):	> 95
Design Life (years):	60
Plant Footprint (m^2):	90 000
Coolant/Moderator:	Light water
Primary Circulation:	Forced circulation
System Pressure (MPa):	15.5
Main Reactivity Control Mechanism:	Control Rod Driving Mechanisms (CRDM), soluble boron and burnable poison
RPV Height (m):	18.5
RPV Diameter (m):	6.5
Coolant Temperature, Core Outlet (°C):	323
Coolant Temperature, Core Inlet (°C):	296
Integral Design:	Yes
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	No
Low-Temp Process Heat:	Yes
Cogeneration Capability:	Yes
Design Configured for Process Heat Applications:	Yes
Passive Safety Features:	Yes
Active Safety Features:	Yes
Fuel Type/Assembly Array:	UO_2 pellet/17x17 square
Fuel Active Length (m):	2
Number of Fuel Assemblies:	57
Fuel Enrichment (%):	< 5
Fuel Burnup (GWd/ton):	60
Fuel Cycle (months):	36
Number of Safety Trains:	4
Emergency Safety Systems:	Active and passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	36
Distinguishing Features:	Coupling with desalination system and process heat application
Modules per Plant:	1
Estimated Construction Schedule (months):	36
Seismic Design (g):	> 0.18 (automatic shutdown)
Predicted Core Damage Frequency (per reactor year):	10^{-6} (for internal events)
Design Status:	Licensed/Certified

Safety Features

The design incorporates engineered safety systems that are designed to function automatically. These consist of a reactor shutdown system, a safety injection system, a passive residual heat removal system (PHRS), a shutdown cooling system and a containment spray system. Additional engineered safety systems include a reactor overpressure protection system and a severe accident mitigation system. The integral arrangement of a reactor vessel assembly fundamentally eliminates the possibility of large loss of coolant accident (LOCA), only small-break LOCA events have a probability of being raised up. As a result the reactor safety is enhanced greatly and the core damage frequency also remarkably reduced.

SMART has engineered safety features that combined active and passive safety features to cope with abnormal operating occurrences and postulated design basis accidents, and severe accidents with core melts. Even if all the electrical power is lost the passive residual heat removal system naturally prevents the reactor core from being overheated. The containment building is resistant to any kind of seismic activity and can withstand possible air-crash incident. These safety features of SMART have been validated through full-scale and scaled performance tests.

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 seawater desalination system may be used in conjunction with the secondary system.

Fuel Characteristics and Fuel Supply Issues

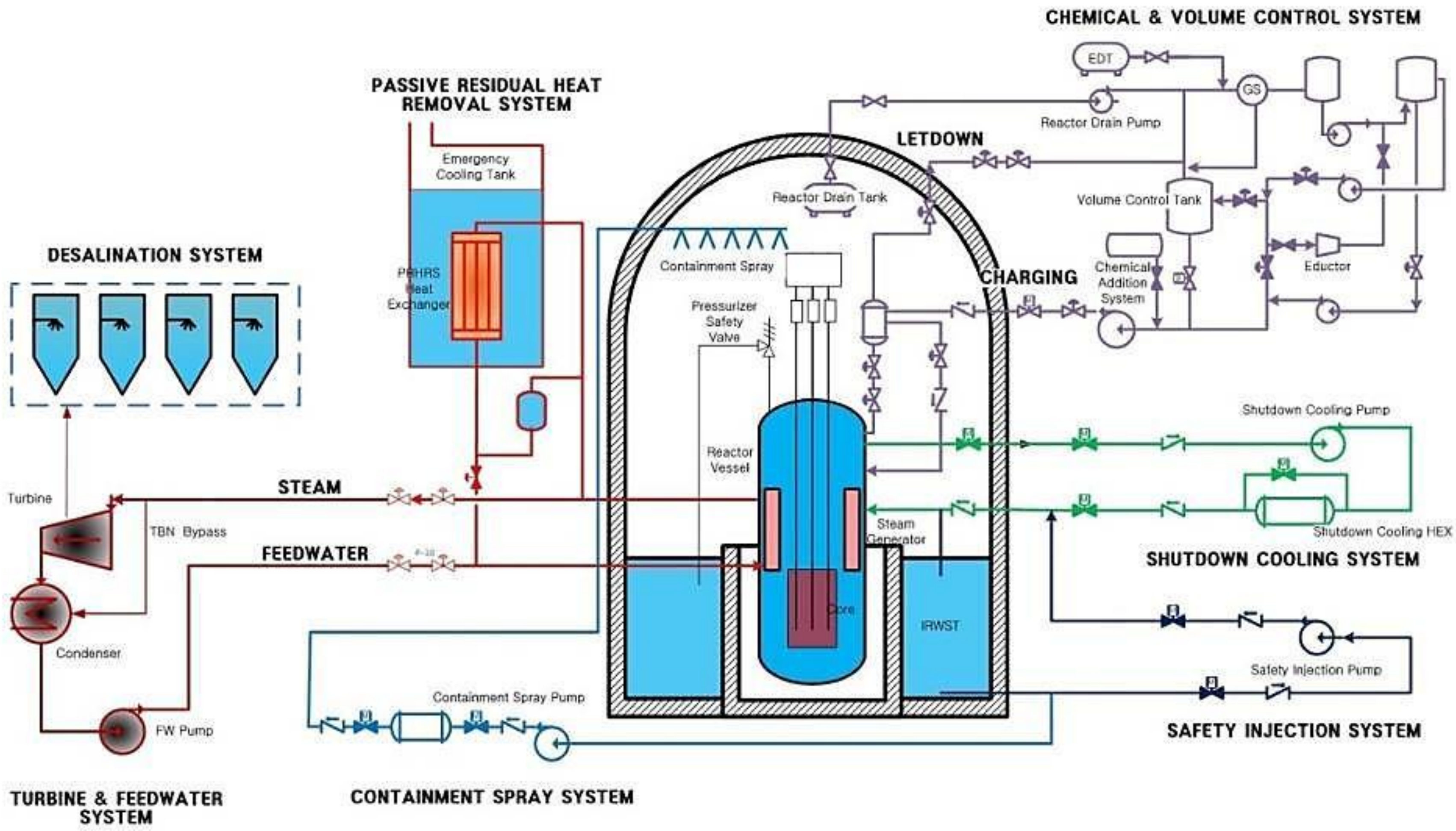
The fuel assembly (FA) is formed by a standard 17x17 square of UO_2 ceramic fuel with the enrichment of less than 5% similar to standard PWR fuel. There are 57 FAs in the core. SMART fuel management is designed to achieve a maximum cycle length between refuelling outages. A simple two-batch refuelling 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. In addition, as for non-proliferation feature, the integral arrangement of reactor components prevents easy access to the fuel as it would require a special equipment to load and unload the fuel in the reactor core. In this case, the reactor and fuel buildings are equipped with a full monitoring system with Closed Circuit Monitoring System (CCTV) to oversee and prevent unauthorized access to the fuel.

Licensing and Certification Status

SMART has been fully licensed in South Korea and standard design of SMART was approved by the Korean Nuclear Safety and Security Commission in July 2012.

R&D of Fully Passive System for SMART

The hybrid (active and passive) safety system currently employed in the current SMART design is planned to be upgraded with fully passive safety system. The passive safety system is being developed to maintain the SMART plant in a safety shutdown condition following design bases accidents such as LOCA and non-LOCA transient events without AC power or operator actions. The passive safety system consists mainly of Passive Safety Injection System (PSIS), Passive Residual Heat Removal System (PRHRS), Passive Containment Cooling System (PCCS), and Automatic Depressurization System (ADS). Based on the current SMART design, the capacity of the PRHRS is increased up to an operation period of at least 72 hours. All the active safety features are being substituted with passive versions, eliminating the necessity for Emergency Diesel Generator (EDG) or operator actions for at least 72 hours period. A program to adopt a fully passive safety system in SMART began from March 2012, and the testing and verification are planned to be completed by the end of 2015.

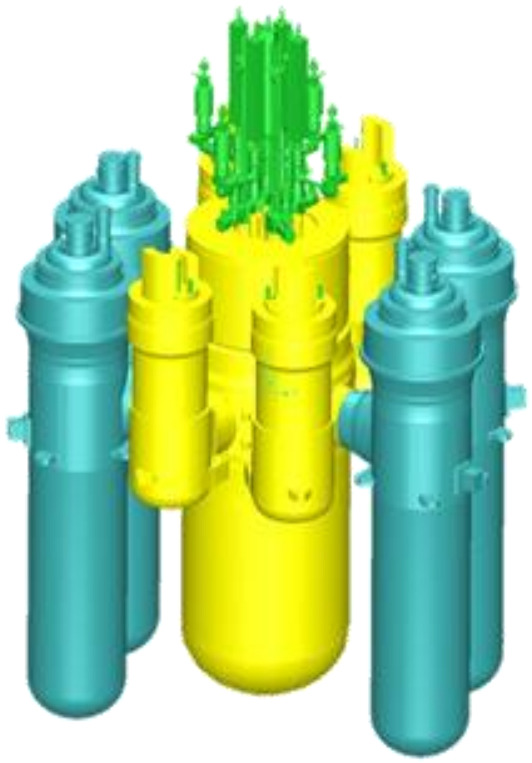


Engineered safety systems (Courtesy of KAERI, with permission)

KLT-40S (OKBM Afrikantov, Russian Federation)

Introduction

The KLT-40S is a PWR developed for a floating nuclear power plant to provide capacity of 35 MW(e) per module. The design is based on the commercial KLT-40 marine propulsion plant and is an advanced version of the reactor providing the long-term operation of nuclear icebreakers under more severe conditions as compared to stationary NPPs. 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. 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. Besides, the floating power unit (FPU) requires 12–15 m deep protected water area with footprint almost 30000 m^2.



Reactor System Configuration of KLT-40S (Courtesy of OKBM Afrikantov, with permission)

Development Milestones

1998	The first project to build a floating nuclear power plant was finished off
2002	After several delays the project was revived by Minatom (Russian Federation Ministry of Nuclear Energy - processor of Rosatom State Corporation)
2002	The environmental impact assessment for KLT-40S reactor systems was approved by the Russian Federation Ministry of Natural Resources
2012	Pevek (Chukotka Autonomous Region) was selected as the site for the installation of small modular NPPs integrated with floating platforms with the KLT-40S reactor system. JSC “Baltiysky Zavod” (Saint Petersburg) was in charge of construction, launching, fitting-out, testing and commissioning the first floating plant named Akademik Lomonosov (2 units with the output of 35 MW(e)
2016~2017	Commercial start

Target Applications

The floating ship-type configuration SMR (KLT-40S) provides cogeneration capabilities for reliable power and heat supply to isolated consumers in remote areas without centralized power supply. Besides, this FPU can be used for seawater desalination complexes as well as for autonomous power supply for sea oil-production platforms.

Specific Design Features

The reactor has a modular design with the core, 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.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	OKBM Afrikantov
Country of Origin:	Russian Federation
Reactor Type:	PWR
Electrical Capacity (MW(e)):	35
Thermal Capacity (MW(th)):	150
Expected Capacity Factor (%):	70
Design Life (years):	40
Plant Footprint (m^2):	30 000
Coolant/Moderator:	Light water
Primary Circulation:	Forced circulation
System Pressure (MPa):	12.7
Main Reactivity Control Mechanism:	Control Rod Driving Mechanism (CRDM)
RPV Height (m):	4.8
RPV Diameter (m):	2.0
Coolant Temperature, Core Outlet (°C):	316
Coolant Temperature, Core Inlet (°C):	280
Integral Design:	No
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	No
Low-Temp Process Heat:	Yes
Cogeneration Capability:	Yes
Design Configured for Process Heat Applications:	Yes
Passive Safety Features:	Partial
Active Safety Features:	Yes
Fuel Type/Assembly Array:	UO_2 pellet/hexagonal
Fuel Active Length (m):	1.2
Number of Fuel Assemblies:	121
Fuel Enrichment (%):	< 20
Fuel Burnup (GWd/ton):	45.4
Fuel Cycle (months):	28
Number of Safety Trains:	2
Emergency Safety Systems:	Active and passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	30
Distinguishing Features:	Floating power unit for heat and electricity
Modules per Plant:	2
Estimated Construction Schedule (months):	48
Seismic Design:	9 point on the MSK scale
Predicted Core Damage Frequency (per reactor year):	0.5×10^{-7}
Design Status:	Under construction, planned commercial start 2016-2017

The steam lines while exiting from the SGs are routed through containment to a set of steam admission valves, and finally into the turbine building for electricity conversion. Cogeneration equipment could be modified into the medium-low temperature heat process concept if one or multiple separation heat exchangers are positioned between the primary and secondary loops.

The KLT-type reactor pressure vessel (RPV) configuration is much comparable with that of VBER-300 design, the difference is that each reactor circulation pump (RCP) is directly hydraulically connected to the RPV. Through acceptance of such a configuration, the core is bottom up cooled by coolant flowing under the pressure.

The Control Rod Drive Mechanism (CRDM) is electric driven and release control and emergency control rods into the core in case of station block-out (SBO). The speed of safety rods in the case of emergency when safety rods are driven by electric motor is 2 mm/s. The average speed of safety rods being driven by gravity is 30~130 mm/s. The containment pressure reduction system is passive-driven and located above the reactor systems.

Possible floating power unit configuration
(Courtesy of OKBM Afrikantov, with permission)

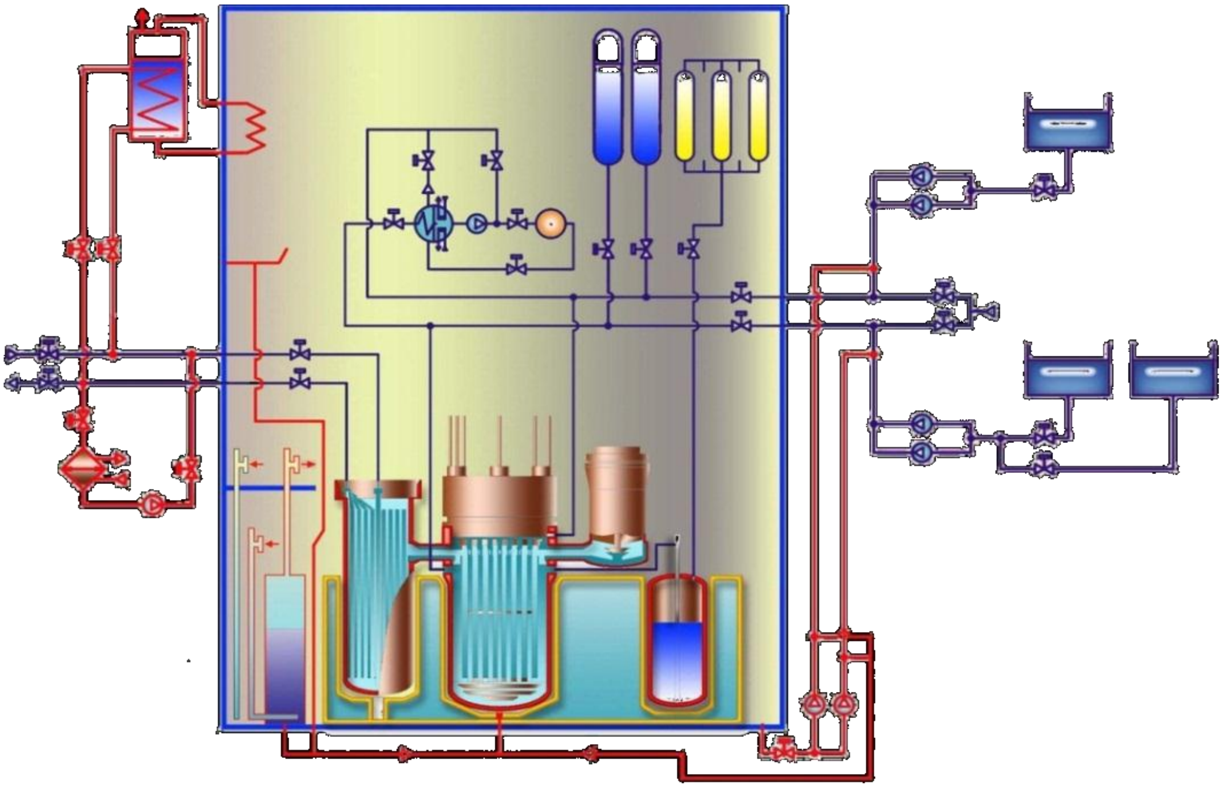
Safety Features

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.

Additional barriers are provided to prevent the release of radioactivity from the FPU caused by severe accidents. Among them are passive and active physically separated and independent safety systems, I&C systems, diagnostic systems, active cooling train through primary circuit purification system's heat exchanger thermally coupled with a 'third' independent circuit exchanging heat energy with ambient sea or lake water, active cooling train through the SGs heat exchangers with decay heat removal accomplished through the condenser which in turn is cooled down by ambient sea or lake water, 2 passive cooling trains through the SGs with decay heat removal via emergency water tank heat exchangers, and venting to atmosphere by evaporation from said tanks.

Both active and passive safety systems are to perform by applying the reactor emergency shutdown, emergency heat removal from the primary circuit, emergency core cooling, radioactive products confinement. The KLT-40S safety concept considers the accident prevention and mitigation system, a physical barriers system, and a system of technical and organizational measures on protection of the barriers and retaining their effectiveness, in conjunction with measures on protection of the personnel, population and environment.

The KLT-40S safety systems installed on FPUs are distinctive from those applied to land-based installations in security of water areas surrounding the FPU, anti-flooding features, anti-collision protection and etc. Passive cooling channels with water tanks and in-built heat exchangers ensure reliable cooling to 24 hours. Actuation of the system is also performed by special devices with passive actuation principle – hydraulically operated pneumatic valves.



KLT-40S flow diagram (Courtesy of OKBM Afrikantov, with permission)

Fuel Characteristics and Fuel Supply Issues

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 FPU based ATES-MM under construction is long term autonomous operation in remote regions with a decentralized power supply. The design requires refuelling after every 3~4 years of operation. In order to maintain a high capacity factor, refuelling is performed 13 days after reactor shutdown when the levels of residual heat releases from spent FAs are still high. Fuel of the reactor systems without site refuelling is delivered to a small-power cogeneration plant importing country as a part of the reactor with the plugged reactor core. The spent nuclear fuel is then stored on board the FPU, and no special maintenance or refuelling ships are necessary.

Single loadings have been accepted in order to provide maximum operation period between refuellings when the fuel is loaded in the core all at once with all fuel assemblies being replaced at the same time. Besides, low-enriched ^{235}U with enrichment below 20% is used to meet non-proliferation requirements.

Licensing and Certification Status

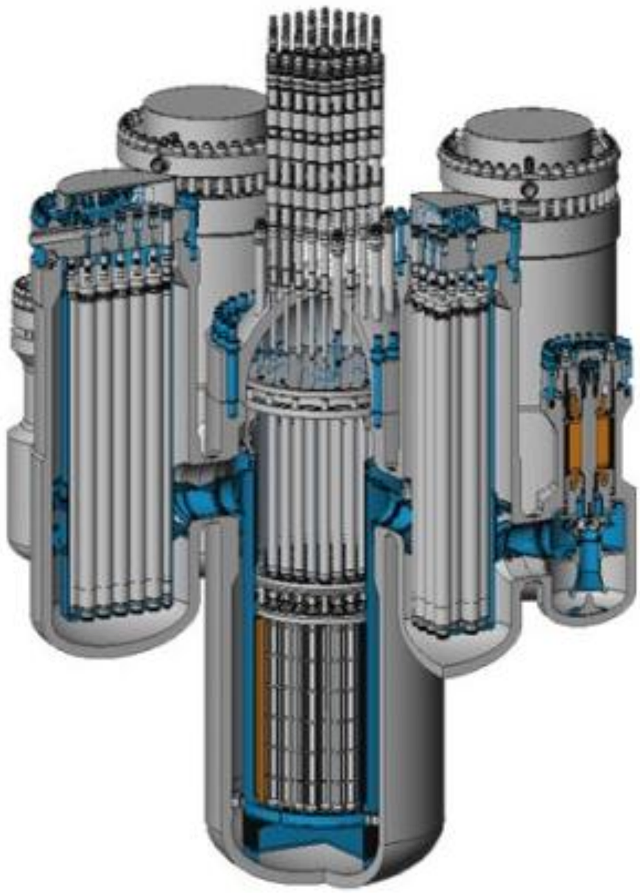
The environmental impact assessment for KLT-40S reactor systems was approved by the Russian Federation Ministry of Natural Resources in 2002. In 2003, the first floating plant using the KLT-40S reactor system received the nuclear site and construction licenses from Rostechnadzor (Russia's nuclear regulator).

The keel of the first FPU carrying the KLT-40S, the Akademik Lomonosov in the Chukotka region, was laid in 2007. The Akademik Lomonosov is to be completed by the end of 2016 and expected electricity production is by 2017.

VBER-300 (OKBM Afrikantov, Russian Federation)

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 and having an electric power output of 325 MW. The VBER-300 design is a result of the evolution of modular marine propulsion reactors. The thermal power increase is reflected in 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 and 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).



Reactor System Configuration of VBER-300 (Courtesy of OKBM Afrikantov, with permission)

Development Milestones

2001	Design activities for the development of VBER reactors
2002	Technical and commercial proposal for the two-unit NPP with VBER-300 Reactor Plant (RP)
2004	Preliminary design of the reactor plant approved by Scientific and Technical Board No.1 and State Nuclear Supervision Body (GosAtomNadzor)
2006	JSC Nuclear Stations (the Kazakhstan-Russian company) was established to promote VBER-300 design
2007-09	Technical assignment for the NPP design and for final designs of the reactor plant, automated process control system and heat-generating plant; Feasibility, Economy and Investment studies for NPP with VBER-300 RP at Mangistaus Region in Kazakhstan
2007-08	Development of the 100-600 MW VBER RP power range
2008-11	Research and development work on the NPP design with VBER-460/600 RP
2011-12	Development of the VBER-600/4 RP based upon the heat exchange loop with increased capacity

Target Applications

The VBER-300 nuclear plants are intended to supply thermal and electrical power to remote areas where centralized power is unavailable, and to substitute capacities of available cogeneration plants on fossil fuels. They are also proposed to be used as power sources for seawater desalination complexes. Nuclear plant with the VBER-300 formed by two reactor units in the steam-condensing mode is capable of generating 600 MW(e), which equivalents to power demands of city with 300000 population.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	OKBM Afrikantov
Country of Origin:	Russian Federation
Reactor Type:	Integral PWR
Electrical Capacity (MW(e)):	325
Thermal Capacity (MW(th)):	917
Expected Capacity Factor (%):	> 90
Design Life (years):	60
Plant Footprint (m^2):	N/A
Coolant/Moderator:	Light water
Primary Circulation:	Forced circulation
System Pressure (MPa):	16.3
Main Reactivity Control Mechanism:	Control Rod Driving Mechanism (CRDM), soluble boron
RPV Height (m):	8.5
RPV Diameter (m):	3.6
Coolant Temperature, Core Outlet (°C):	328
Coolant Temperature, Core Inlet (°C):	292
Integral Design:	No
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	No
Low-Temp Process Heat:	Yes
Cogeneration Capability:	Yes
Design Configured for Process Heat Applications:	Yes
Passive Safety Features:	Yes
Active Safety Features:	Yes
Fuel Type/Assembly Array:	UO_2 pellet/hexagonal
Fuel Active Length (m):	3.5
Number of Fuel Assemblies:	85
Fuel Enrichment (%):	4.95
Fuel Burnup (GWd/ton):	50
Fuel Cycle (months):	72
Number of Safety Trains:	2
Emergency Safety Systems:	Active and passive
Residual Heat Removal Systems:	Active and passive
Refuelling Outage (days):	N/A
Distinguishing Features:	Power source for floating nuclear power plants
Modules per Plant:	1
Estimated Construction Schedule (months):	N/A
Seismic Design (g):	0.25
Predicted Core Damage Frequency (per reactor year):	10^{-6}
Design Status:	Licensing stage

Specific Design Features

This compact VBER-300 reactor system comprises of the steam generating system in a restricted space of the reactor compartment, enhanced system reliability and long design life and refuelling cycle characteristics. The main technical characteristics of VBER RP design are modularity, once-through steam generators with titanium tube system, canned main circulating pumps, low density core and optimal combination of active and passive safety systems.

In this design, all components of the primary loop are directly connected to the RPV, excluding pressurizer. 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.

Besides focus on land-based installations the VBER-300 design can also be configured as a FPU and can be arranged to operate individually or as a set of modules increasing the thermal power output by the means of scaling up the equipment and remaining the same reactor system configuration.

The basic architectural solution for the land-based power unit is to set the reactor system (including its servicing systems, a spent fuel pool, and auxiliaries) in a double protective resistant to the air-crash. The inner steel containment, 36 m in diameter and 49 m height, condenses the steam generated out of medium to large size LOCAs and an outer concrete structure of 44 m height is serving as a protection against natural and man caused impacts.

Floating Power Unit Features

The VBER-300 floating nuclear power unit is a non-self-propelled autonomous floating structure related to the pillar-class ships according to Russian Sea Navigation Register classification. The floating power unit is located on a platform consisting of three pontoons (one central and two peripheral ones). The independent reactor plants are in-line located in the central pontoon. Each reactor plant consists of the following divisions: reactor division, power plant control board and electrical equipment divisions; areas for the plant refuelling and repair are also provided. A steel protective shell (containment system) houses main equipment of the reactor module with related service systems. A fuel storage is located in the central pontoon between the reactor plants. Turbine generators and their related equipment and systems are located in stern and fore sections of the central pontoon. Electrical equipment for power transformation, distribution and supply (at up to 220 kV voltage) to coastal objects and that for power supply to the plant house loads are located in a left board pontoon. A composite steel-ferro concrete vessel of the power unit eliminates the need for scheduled docking during the plant life time. Period between overhauls (20 years.) is dictated by key reactor plant equipment life that is about 150000 h. The PAES-600 NPP operating life is 60 years.

Safety Features

The safety assurance and 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. Emergency RHRS by means of passive cooling channels with water tanks and in-built heat exchangers ensure reliable cooling to 72 hours and more. System actuation is performed by means of passive actuation principle – hydraulically operated pneumatic valves.

The system of pressure decrease in the containment utilized to prevent containment damage and reduce radioactivity in case of design and beyond the design basis accidents. The estimated reactor unit strength holds up against seismic impacts of maximum magnitude of 0.25 g.

Accidents with small and medium primary coolant leakages are prevented by the elimination of sprinkler system, low-pressure emergency injection system and core passive flooding system.

Fuel Characteristics and Fuel Supply Issues

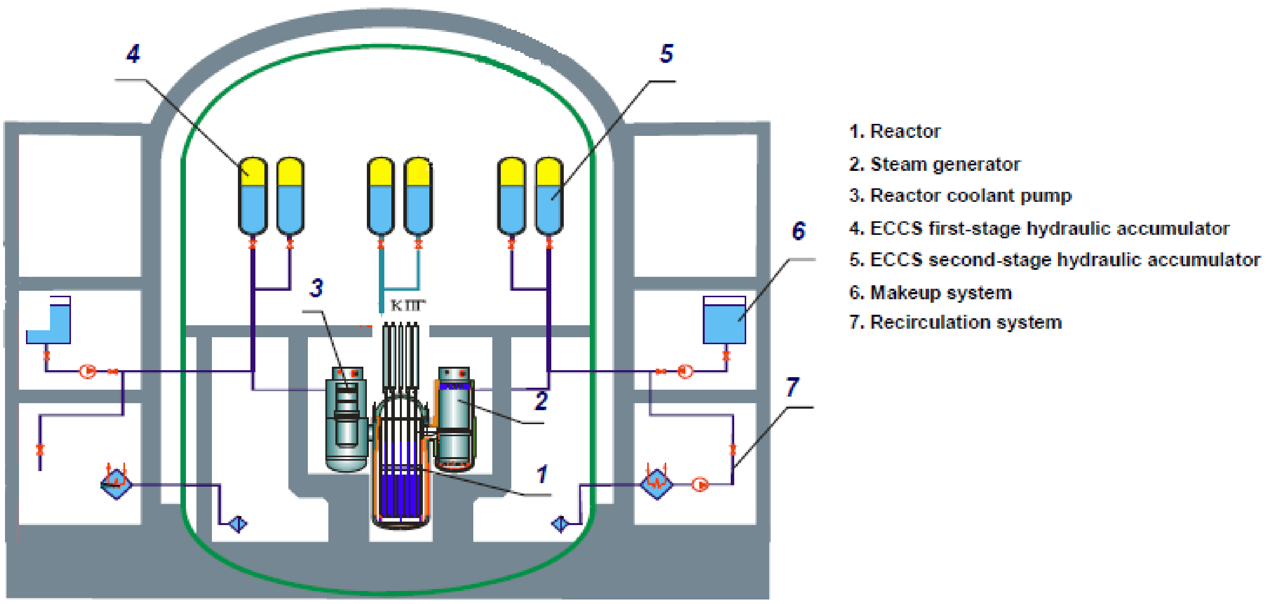
The reactor core comprises of 85 hexagonal fuel assemblies (FAs) which are placed in the reactor cavity in nods of a regular triangular lattice with a space interval of 236 mm. The VBER-300 design concept allows a flexible fuel cycle for the reactor core with standard WWER fuel assemblies (FAs). The 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.

Licensing and Certification Status

The VBER-300 preliminary design was completed in 2002, and a technical and commercial proposal (a shorter version of technical and economic investigation) for construction of a land based or floating 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.

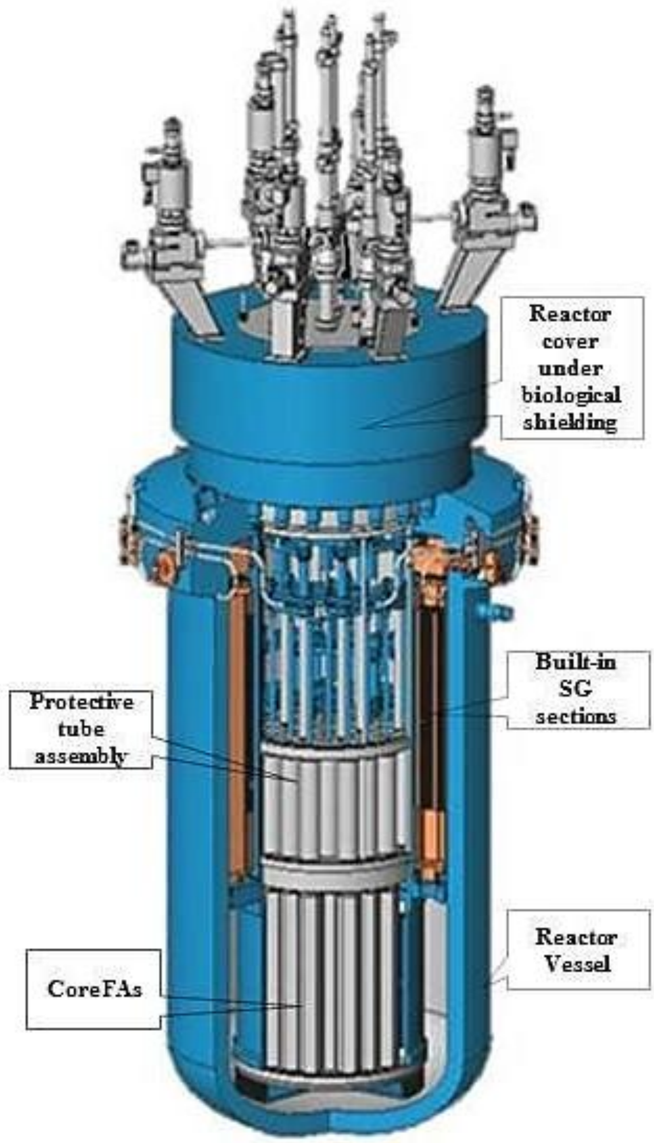


Emergency Core Cooling System
(Courtesy of OKBM Afrikantov, with permission)

ABV-6M (OKBM Afrikantov, Russian Federation)

Introduction

The ABV-6M installation is a nuclear steam generating plant producing 14 MW(th) or 6 MW(e) in cogeneration mode. This integral PWR has a 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 source providing easy transport to the site, rapid assembly and safe operation during 10 – 12 years without refuelling at the berthing platform or on the coast. Plant maintenance and repair, refuelling and nuclear waste removal are fulfilled at special enterprises suitable for that purpose.



Reactor System Configuration of ABV-6M (Courtesy of OKBM Afrikantov, with permission)

Development Milestones

1993	The final design was developed for the prototype reactor plant and Volnolom floating NPP
2006	The feasibility study is developed for construction of the floating NPP with ABV-6M for the Far North (settlement Tiksi, settlement Ust-Kamchatsk)
2007	The feasibility study is developed for construction of the floating NPP with ABV-6M for Kazakhstan (City of Kurchatov)
2014	The final design is being developed for a transportable reactor plant under the contract with Minpromtorg (RF Ministry of Industry and Trade)

Target Applications

The ABV-6M 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 115 m, a beam of 26 m, a draft of 3.5 m and a displacement of 8000 t. Depending on the needs of the region, the floating nuclear power plant can generate electric power or provide heat and power cogeneration or can be used for other applications. Besides, a land-based configuration of the plant is also applicable.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	OKBM Afrikantov
Country of Origin:	Russian Federation
Reactor Type:	PWR
Electrical Capacity (MW(e)):	6
Thermal Capacity (MW(th)):	38
Expected Capacity Factor (%):	> 70
Design Life (years):	40
Plant Footprint (m^2):	N/A
Coolant/Moderator:	Light water
Primary Circulation:	Natural circulation
System Pressure (MPa):	16.2
Main Reactivity Control Mechanism:	Control Rod Driving Mechanism (CRDM)
RPV Height (m):	6
RPV Diameter (m):	2.4
Coolant Temperature, Core Outlet (°C):	325
Coolant Temperature, Core Inlet (°C):	250
Integral Design:	Yes
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	No
Low-Temp Process Heat:	Yes
Cogeneration Capability:	Yes
Design Configured for Process Heat Applications:	Yes
Passive Safety Features:	Yes
Active Safety Features:	Yes
Fuel Type/Assembly Array:	UO_2/hexagonal
Fuel Active Length (m):	900
Number of Fuel Assemblies:	121
Fuel Enrichment (%):	19.7
Fuel Burnup (GWd/ton):	N/A
Fuel Cycle (months):	120-144
Number of Safety Trains:	2
Emergency Safety Systems:	Passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	N/A
Distinguishing Features:	Natural circulation in the primary circuit for land based and floating nuclear power plants
Modules per Plant:	1
Estimated Construction Schedule (months):	N/A
Seismic Design:	N/A
Predicted Core Damage Frequency (per reactor year):	10^{-6}
Design Status:	Detailed design

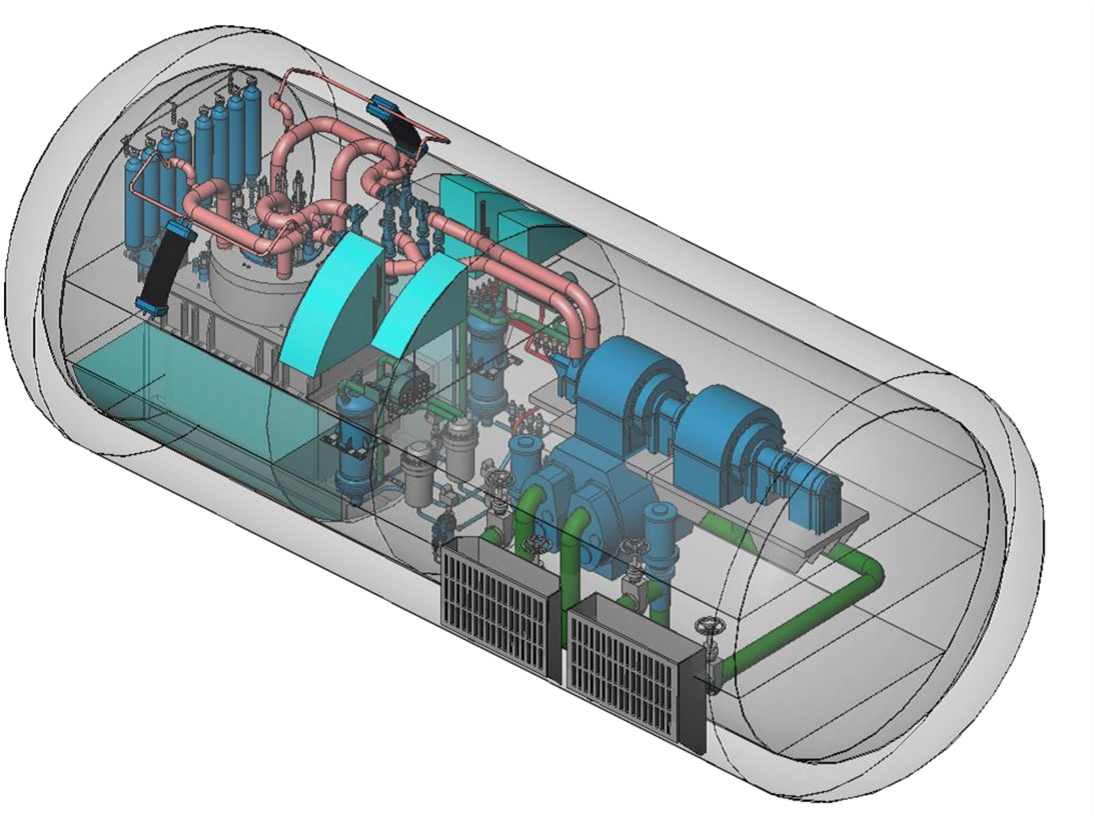
Specific Design Features

The basic design decisions for other equipment units are also selected considering the optimization of the capital and operation costs. Land-based and floating power units use design solutions for minimizing mass and overall dimensions, feasibility to deliver steam generating units (SGU) or Reactor Pressure Vessel (RPV) in a containment vessel by railway or water transport at minimal scope of installation activities at the manufacturing plant or at the construction site. Structural optimizations of main equipment allows for delivery of the NPP at the basing site through water with minimal technological issues. The ABV-6M is particularly applicable for modular-transportable power units, and floating power units located in sea shallow water areas also in the rivers. Therefore, single-reactor power units are the most preferable designs. From the economic point of view it reduces the startup capital costs, construction and payback periods. For the floating power units it means significant reduction of displacement and feasibility to locate them in the sea shallow water areas and in the rivers.

Land-Based Modular-Transportable NPP

The stationary nuclear power plant — land based — of the ABV-6M is fabricated as large, ready-made units; these units are transported to the site by water. The main module consists of reactor plant, steam-turbine plant, part of electrical plant and control systems. It has a mass of 3000 t and is 44 m in length and 10 m in diameter.

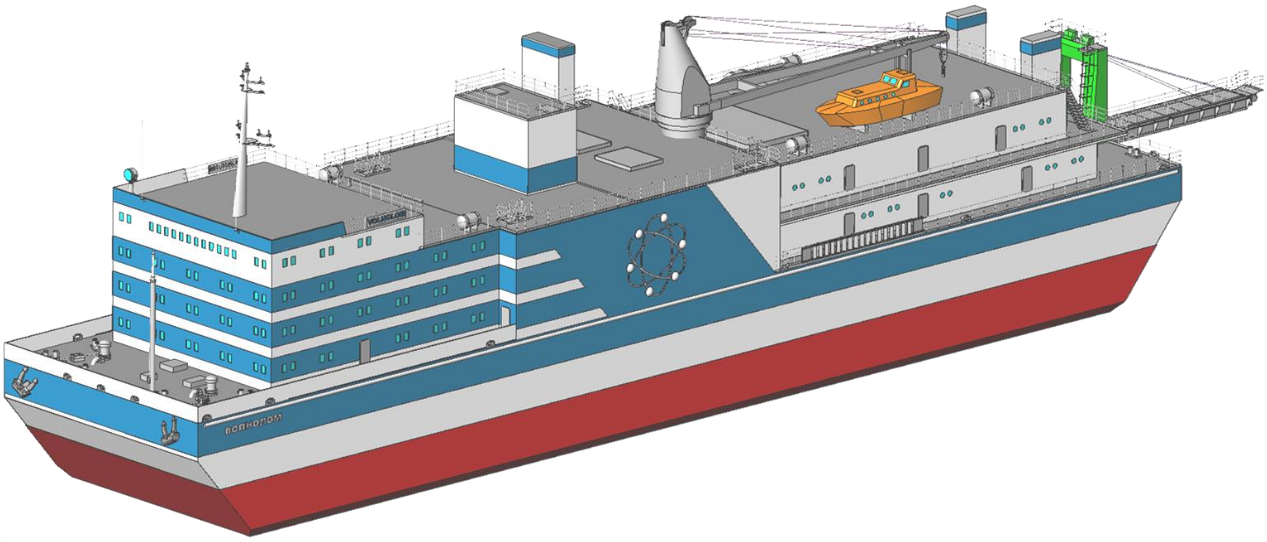
Due to modularization of the main equipment into large–size prefabricated modules the scope of construction and installation activities to be performed at the installation site has been significantly reduced. The delivered modules are installed on the prepared foundation, interconnected by appropriate pipelines, and then easy-to-install modular structures are assembled, providing protection of the power unit against the external factors.



The conceptual view of the power unit (Courtesy of OKBM Afrikantov, with permission)

Floating Power Unit Features

The floating nuclear power plant is factory fabricated. The key distinguishing characteristic of ABV-6M reactor from KLT-40S, is that it can be transported through the estuary of a river with only 3.5 m deep, whenever it is necessary.



The conceptual view of the floating power unit
(Courtesy of OKBM Afrikantov, with permission)

The reactor operates under the normal PWR conditions of 16.2 MPa in the reactor pressure vessel. The steam generators located inside the vessel generate 295°C steam at 3.83 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.

Safety Features

Traditionally, while designing, priority is given to the safety of the reactor unit under development. This is especially vital considering that it is located close to a settlement and at the same time it is far from the main technical bases, which could provide timely technical support. In view of its small power as compared even with the operating transportation plants of nuclear ships, the majority of emergency processes are milder and often do not require active systems performance.

Land-based and floating power units use the advanced active and passive safety systems and design solutions for emergency cooling over an unlimited time during design-basis and beyond design-basis accidents. Also, design decisions aimed at coolant retention during LOCA.

Low thermal capacity of accepted RP allows using natural circulation at primary coolant circuit and passive safety systems as primary safety systems. The autoprotective features of the NPP under conditions of the hard-to-reach territories in the Far North, Siberia, the Far East and the Arctic region were improved.

Fuel Characteristics and Fuel Supply Issues

The core comprises of 121 hexagonal fuel assemblies of cassette type with active part height of 900 mm, similar to the FAs in KLT-40S the RP is used to reduce financial expenses and time. The core is designed to operate for 10 – 12 years. To provide export potential cermet fuel is used with uranium-235 enrichment less than 20%. This fuel composition was proved during creation of KLT-40S RP for FPU *Academic Lomonosov*, however this innovative fuel composition has not yet been properly mastered in production and that determines it high prime cost. FPU or main modules of land-based NPP are transported to special enterprise for repair and refuelling.

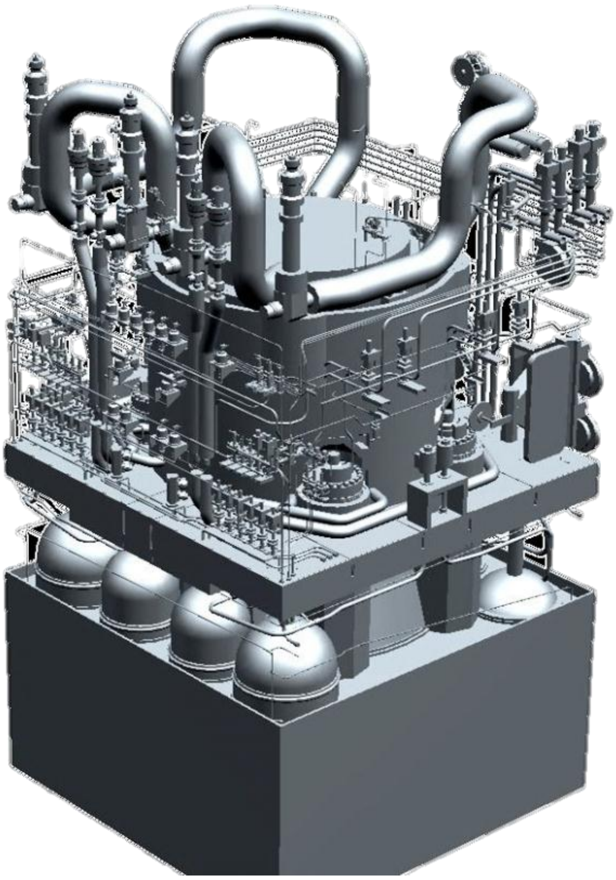
Licensing and Certification Status

Currently, the development of ABV-6M is at the finishing stage.

RITM-200 (OKBM Afrikantov, Russian Federation)

Introduction

The RITM-200 is being designed by OKBM Afrikantov as an integral reactor with forced circulation for use of the multipurpose nuclear icebreaker, floating and land-based NPPs with an electrical output of 50 MW(e). The main design approach is an efficient combination of passive and active safety means and systems, optimal use of the normal operational and safety systems. The structure of the RPV was designed by incorporating the experience gained from the previous plant generation, requirements of the up-to-date safety norms, and reduction of liquid waste.



Reactor System Configuration of RITM-200
(Courtesy of OKBM Afrikantov, with permission)

Development Milestones

2009	Detailed design of the reactor plant was finished
2012	Equipment construction start
2016	RPV complete delivery
2017	Commercial start

Target Applications

The RITM-200 is designed to provide the 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.

Specific Design Features

The RITM-200 design employs a low enriched cassette type reactor core similar in design to the KLT-40S. The fuel enrichment is up to 20% 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. Each of the four SGs have three 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 pressurizing system. The SGs produce steam of 295°C at 3.82 MPa flowing at 248 t/h.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	OKBM Afrikantov
Country of Origin:	Russian Federation
Reactor Type:	Integral PWR
Electrical Capacity (MW(e)):	50
Thermal Capacity (MW(th)):	175
Expected Capacity Factor (%):	65
Design Life (years):	40
Plant Footprint (m^2):	N/A
Coolant/Moderator:	Light water
Primary Circulation:	Forced circulation
System Pressure (MPa):	15.7
Main Reactivity Control Mechanism:	Control Rod Driving Mechanism (CDRM)
RPV Height (m):	8.5
RPV Diameter (m):	3.3
Coolant Temperature, Core Outlet (°C):	313
Coolant Temperature, Core Inlet (°C):	277
Integral Design:	Yes
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	No
Low-Temp Process Heat:	No
Cogeneration Capability:	No
Design Configured for Process Heat Applications:	Yes
Passive Safety Features:	Yes
Active Safety Features:	Yes
Fuel Type/Assembly Array:	UO_2 pellet/hexagonal
Fuel Active Length (m):	1.2/1.65
Number of Fuel Assemblies:	199
Fuel Enrichment (%):	< 20
Fuel Burnup (GWd/ton):	68.4/51.2
Fuel Cycle (months):	54/84
Number of Safety Trains:	2
Emergency Safety Systems:	Active and passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	40
Distinguishing Features:	Developed for universal nuclear ice breakers
Modules per Plant:	2
Estimated Construction Schedule (months):	48
Seismic Design (g):	3
Predicted Core Damage Frequency (per reactor year):	0.9×10^{-6}
Design Status:	Under construction, planned commercial start 2017

The core is designed to operate for 4,5/7 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 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.

Safety Features

The main design approach is a reasonable combination of passive and active safety means and systems, optimal use of the normal operational and safety systems, includes:

- passive pressure reduction and cooling down systems are introduced (efficiency of the systems is confirmed by bench testing);
- pressure compensation system that is divided into two independent groups to minimize diameter of coolant leak;
- main circulation path of the primary circuit is located in a single vessel;
- header scheme of primary coolant circulation is introduced, which ensures advanced vitality of the plant during SG and MCP failures.

The safety concept of the RITM-200 reactor plant is based on the defence-in-depth principle combined with the plant self-protection and use of passive systems. Properties of intrinsic self-protection are intended for power density self-limitation and reactor self-shutdown, limitation of primary coolant pressure and temperature, heating rate, primary circuit depressurization scope and outflow rate, fuel damage scope, maintaining of reactor vessel integrity in severe accidents and form the image of a “passive reactor”, resistant for all possible disturbances.

Emergency cooling down systems (ECDS) are intended to remove residual heat from the reactor core during emergencies caused by heat removal failure and depressurization of the primary circuit. In particular, a combined passive ECDS is being developed. It is designed as a combination of several independent channels connected to different SGs. Each channel contains an emergency cooling down tank; a heat exchanger which is located in the ECD tank; a water storage tank, and air cooled heat exchanger. Thus, the system uses two types of heat exchangers: the ones cooled with water, similar to those accepted in KLT-40S RP and RITM-200 designs, and heat exchangers cooled with atmospheric air. At the initial stage heat is removed through two heat-exchangers: the air goes directly to atmospheric air, and the water goes to water of ECD tank, what allows ensuring peak heat removal. After evaporation of water in the ECD tank, the air heat exchanger still operates and provides RP further cooling down during unlimited period. This combination allows providing minimal overall dimensions of the air heat–exchanger and water volume in the ECD tank.

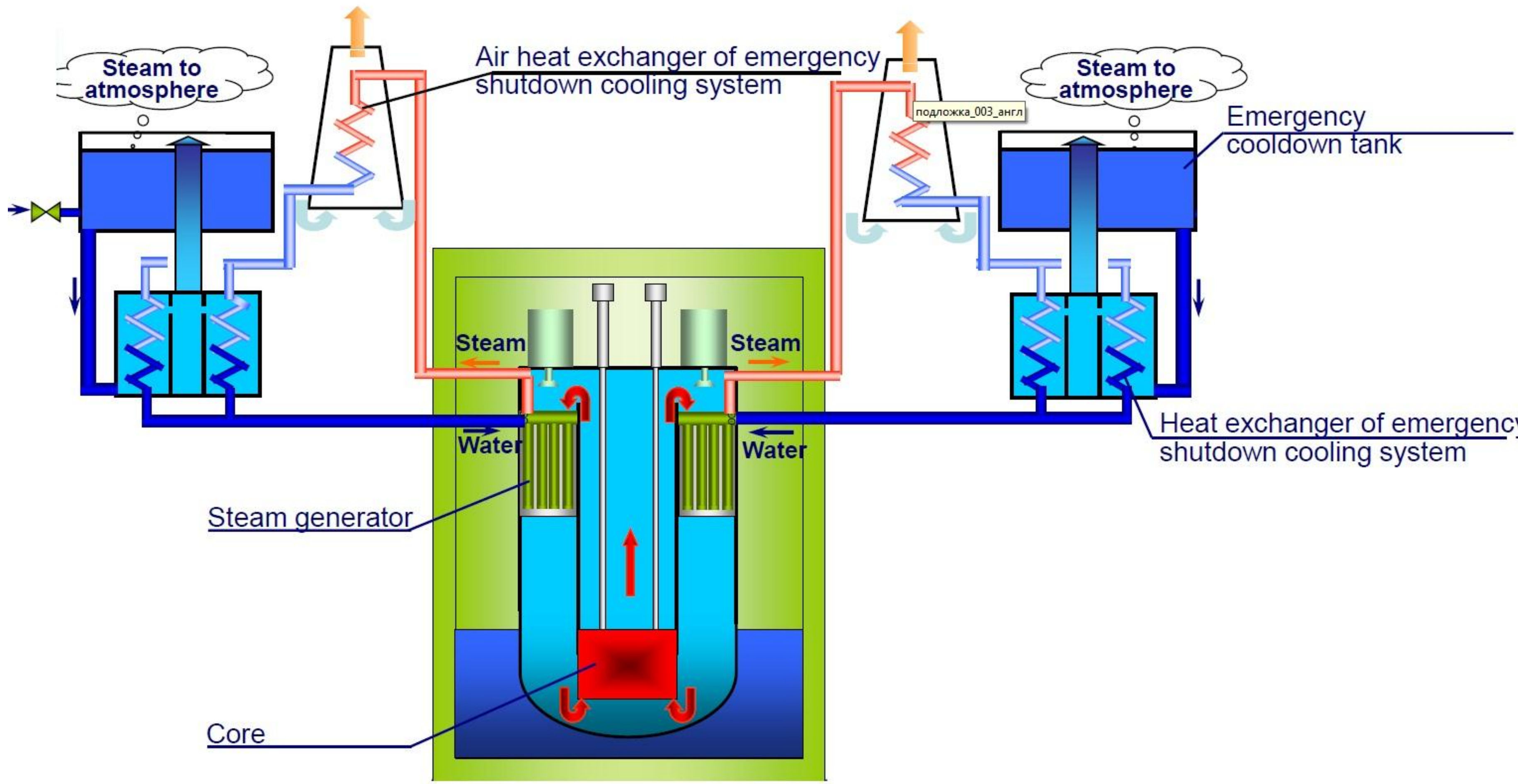
In case of a failure of automated systems, self-actuating devices are installed to actuate directly under the medium pressure of the primary circuit, they ensure tripping of the reactor and initiate the safety systems.

The CRDM is electrically driven and releases control and emergency control rods into the core in case of station block-out (SBO). The speed of safety rods in the case of emergency when safety rods are driven by electric motor is 2 mm/s. The average speed of safety rods being driven by gravity is 30 to 130 mm/s. The containment pressure reduction system is passive-driven and provides localization of steam-water mixture during LOCA in the containment of the damaged RP, application of self-actuating devices: hydraulically operated pneumatic valves (at increase of pressure in containment), pressure-actuated contact breakers.

Licensing and Certification Status

The RP designs were developed in conformity with Russian laws, norms and rules for ship nuclear power plants and safety principles developed by the world community and IAEA recommendations.

The current status of RITM-200 concept is as two reactor plants for the first multipurpose icebreaker (complete delivery in 2016) are being manufactured. Serial deliveries of reactor units for two consequent nuclear ice-breakers will be in 2017 and 2018.

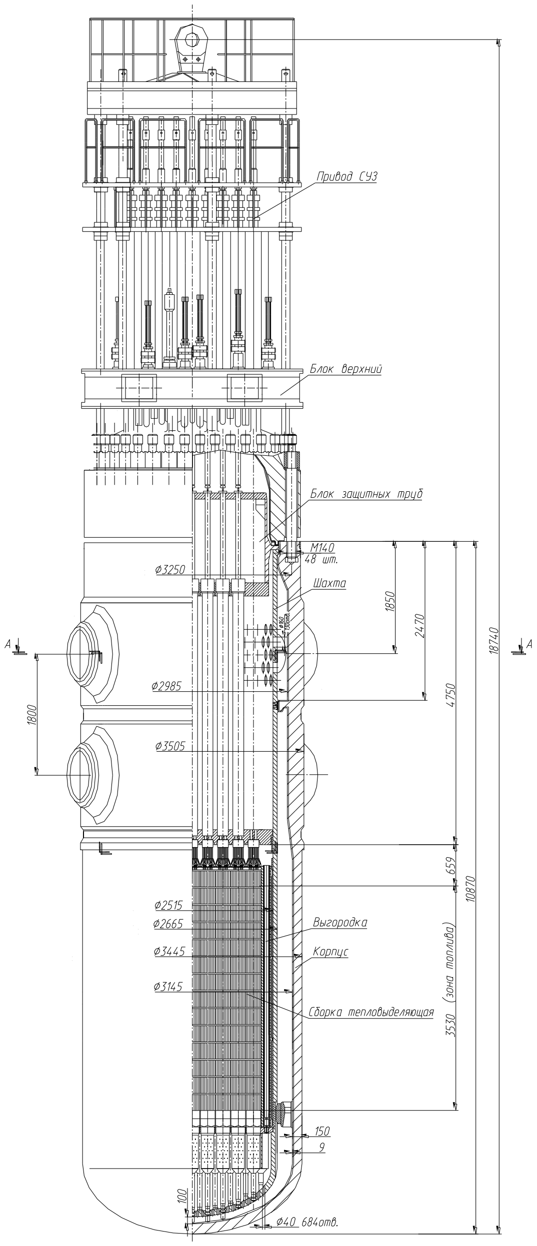


Emergency Core Cooling Down System (Courtesy of OKBM Afrikantov, with permission)

VVER-300 (OKB Gidropress, Russian Federation)

Introduction

The VVER-300 is designed to generate a thermal power of 850 MW(th) or an electrical power of about 300 MW(e). 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.



Reactor System Configuration of VVER-300 (Courtesy of OKB Gidropress, with permission)

Development Milestones

2008	Conceptual Design Development

Target Applications

The VVER-300 design is to be deployed in the remote areas with power grids of limited capacity.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	OKB Gidropress
Country of Origin:	Russian Federation
Reactor Type:	WWER
Electrical Capacity (MW(e)):	300
Thermal Capacity (MW(th)):	850
Expected Capacity Factor (%):	> 90
Design Life (years):	60
Plant Footprint (m^2):	~900 (containment building)
Coolant/Moderator:	Water/ Water
Primary Circulation:	Forced circulation
System Pressure (MPa):	16.2
Main Reactivity Control Mechanism:	Mechanical control and protection system (CPS), soluble boron
RPV Height (m):	10.870
RPV Diameter (m):	3.1
Coolant Temperature, Core Outlet (°C):	325
Coolant Temperature, Core Inlet (°C):	295
Integral Design:	No
Power Conversion Process:	Rankine Cycle
High-Temp Process Heat:	No
Low-Temp Process Heat:	Yes
Cogeneration Capability:	Yes
Design Configured for Process Heat Applications:	Yes
Passive Safety Features:	Yes
Active Safety Features:	Yes
Fuel Type/Assembly Array:	UO_2 pellets/hexagonal (U-Pu)
Fuel Active Length (m):	3.53 (cool)
Number of Fuel Assemblies:	85
Fuel Enrichment (%):	3.3 – 4.79
Fuel Burnup (GWd/ton):	38 – 65
Fuel Cycle (months):	18 – 24
Number of Safety Trains:	2 – 4
Emergency Safety Systems:	Active and Passive
Residual Heat Removal Systems:	Active and Passive
Refuelling Outage (days):	N/A
Distinguishing Features:	Based on the operating experience of various WWER type reactor designs
Modules per Plant:	1
Estimated Construction Schedule (months):	48
Seismic Design:	VIII-MSK 64
Predicted Core Damage Frequency (per reactor year):	10^{-6}
Design Status:	Conceptual design development

Specific Design Features

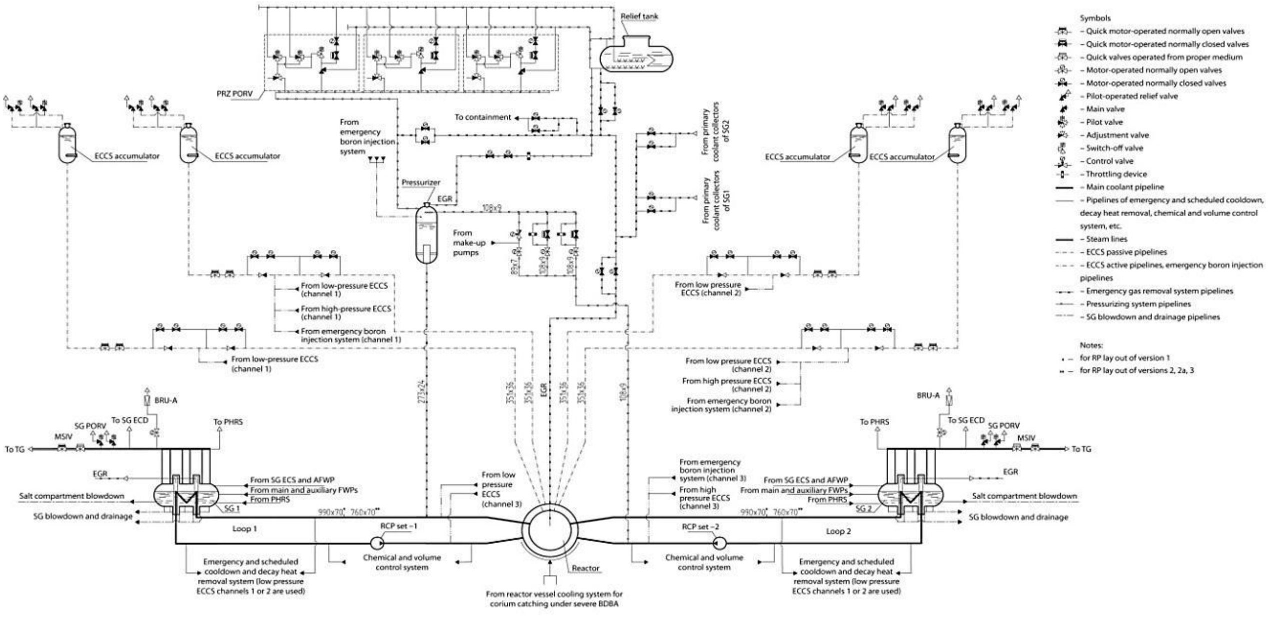
The reactor is a vertical high-pressure vessel with reactor top head and comprising internals (protective tube unit, core barrel, core baffle), core, control rods and in-core instrumentation detectors. Drive mechanism housings are installed on top head nozzles (CPS nozzles). Electromagnet units proposed for CPS control rod core axial displacement are fastened outside the housings. The reactor is housed inside the concrete cavity provided with biological and thermal shields and the cooling system. The reactor vessel is supported with a supporting shoulder on the supporting ring, and fastened in the supporting truss. The reactor is prevented from lateral displacements by the thrust ring installed on the vessel flange. Reactor fastening in the concrete cavity at two levels provides safe prevention from displacements at seismic impacts and pipeline breaks. Cooling of concrete cavity, electric equipment, in-core instrumentation nozzles and drives is provided with air.

The reactor primary loop consists of 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 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 deaerator plant.

The layout of the commercial reactor plant V-320 is used as reference for development of main equipment layout in reactor plant V-478. The only difference is that the number of circulation loops (and, respectively, the number of SG and RCP sets) is decreased from four to two.

The reactor plant is housed within the double leak-tight reinforced-concrete containment of cylindrical shape with top restriction of hemispherical vault. The containment internal wall is also considered as a support for a polar crane.



Reactor core flow diagram (Courtesy of OKB Gidropress, with permission)

Safety Features

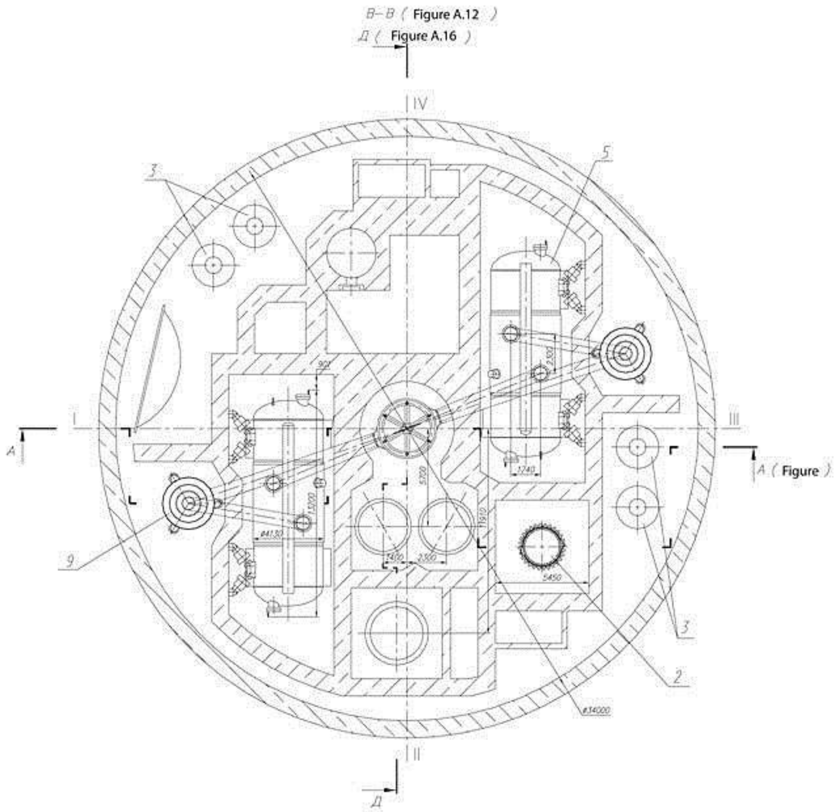
The design was developed according to the requirements of the Russian Federation's current regulations, atomic energy standards, IAEA safety regulations and other recommendations, as well as the requirements of European operators of NPPs. The safety features used in the V-478 design include: SG emergency cooldown systems, emergency gas removal systems, 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 strategy of design basis accidents elimination is based both on active and passive safety systems. The strategy of beyond design-basis accidents is mainly based on passive safety systems and beyond design-basis accident control systems.

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 adopted 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.

The main steam lines disconnection system is proposed for SG fast and reliable isolation from leak. The system of main steam lines disconnection is proposed for operation under all accident conditions requiring SG isolation like the break of steam lines from SG to turbine stop valves in SG isolated and non-isolated parts, the break of feed lines in the section from SG to the back pressure valve and primary-to-secondary leaks.

The NPP ensures safety at seismic impacts to safe shutdown earthquake (SSE) inclusively and generation of electrical and thermal energy to operating basis earthquake (OBE) level inclusively according to the requirements of Design Standards for Earthquake-Proof Nuclear Power Plants.



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

Fuel Characteristics and Fuel Supply Issues

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.

Licensing and Certification Status

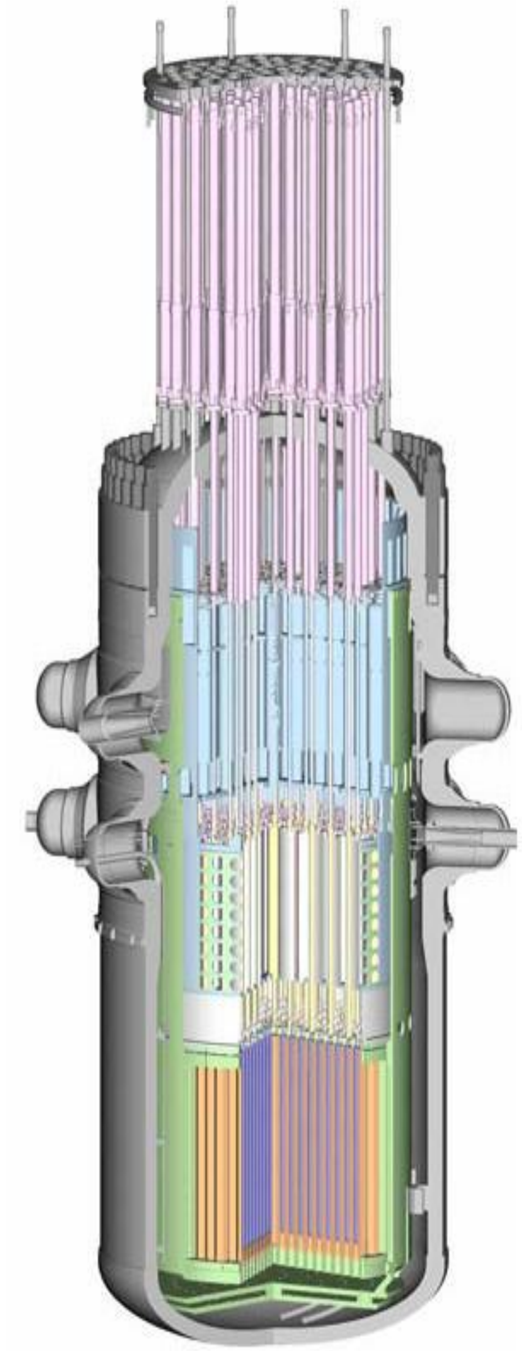
The composition and design of the main components, equipment and systems are based on existing designs, improved according to the up to date requirements that enable improved performance and ensure the required safety level. Basic technical solutions for the nuclear power plant are proved by the more 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 years.

VK-300 (RDIPE, Russian Federation)

Introduction

The VK-300 is a 750 MW(th) or 250 MW(e) simplified integral BWR with natural circulation of coolant and passive systems. The VK-50 SBWR reactor operated in the Russian Federation for 50 years and has been used as a prototype for the VK-300 design. To enhance safety and reliability, the design configuration has incorporated inherent safety features and passive safety systems. The design aim is to achieve improvement in the economics through system simplification.



Reactor Configuration of VK-300
(Courtesy of RDIPE, with permission)

Development Milestones

1998	Conceptual design development
2002	Detailed design development
2003	Cogeneration plant conceptual design development
2004	Feasibility study of the pilot cogeneration plant
2009	Feasibility study of the pilot cogeneration plant upgrade
2013	Design validation, actualization and commercialization

Target Applications

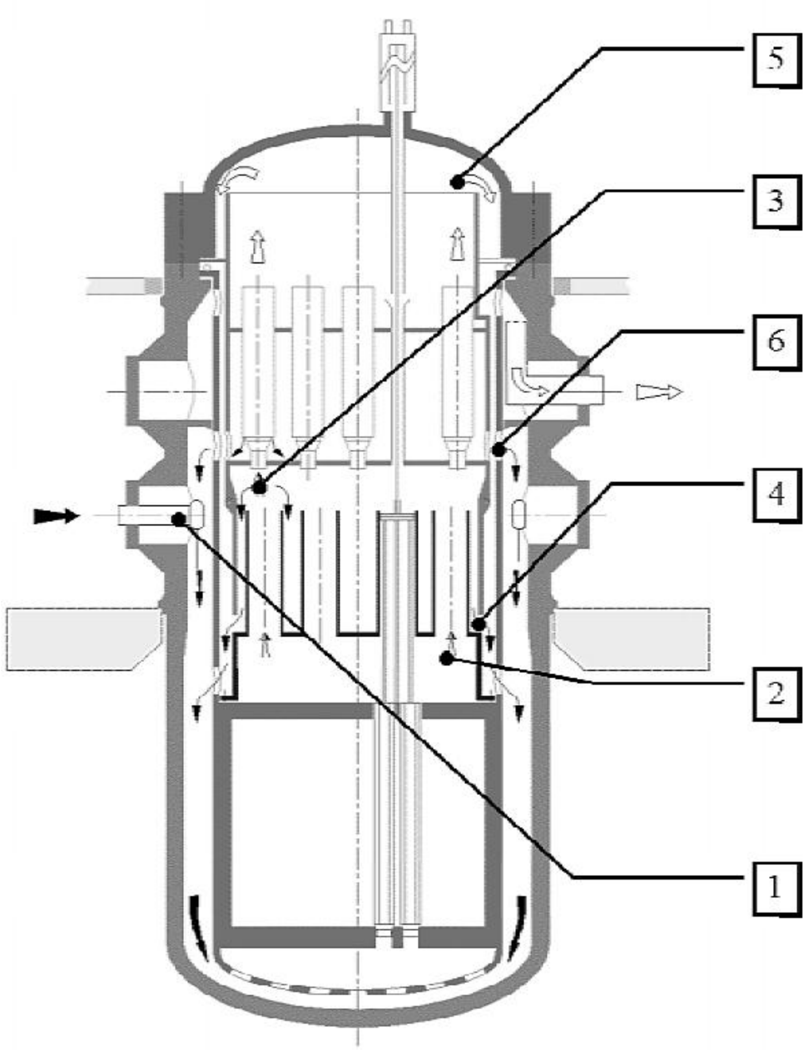
VK-300 reactor facility specially oriented to effective cogeneration of electricity and heat for district heating and for seawater desalination having excellent characteristics of safety and economics.

Specific Design Features

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.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	Research and Development Institute of Power Engineering, (RDIPE)
Country of Origin:	Russian Federation
Reactor Type:	Simplified Integral BWR
Electrical Capacity (MW(e)):	250
Thermal Capacity (MW(th)):	750
Expected Capacity Factor:	0.93
Design Life (years):	60
Plant Footprint (m^2):	40000
Coolant/Moderator:	Light water
Primary Circulation:	Natural circulation
System Pressure:	6.9 MPa
Main Reactivity Control Mechanism:	Rod insertion
RPV Height (m):	13100
RPV Diameter (m):	4.535
Coolant Temperature, Core Outlet (°C):	285
Coolant Temperature, Core Inlet (°C):	190
Integral Design:	Yes
Power Conversion Process:	Direct Rankine Cycle
High-Temp Process Heat:	No
Low-Temp Process Heat:	Yes
Cogeneration Capability:	Yes
Design Configured for Process Heat Applications:	Yes
Passive Safety Features:	Yes
Active Safety Features:	No
Fuel Type/Assembly Array:	UO_2 pellets/Hexahedron
Fuel Active Length (m):	2.42
Number of Fuel Assemblies:	313
Fuel Enrichment:	4%
Fuel Burnup (GWd/ton):	41.4
Fuel Cycle (months):	72
Emergency Safety Systems:	Passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	17.6
Distinguishing Features:	Innovative passive SBWR based on operating prototype and well developed equipment
Modules per Plant:	1+1
Estimated Construction Schedule (months):	48
Seismic Design:	Max 8 Point of MSK-64
Predicted Core Damage Frequency (per reactor year):	10^{-7}
Design Status:	Detailed design of Reactor and Cogeneration Plant Standard Design



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.

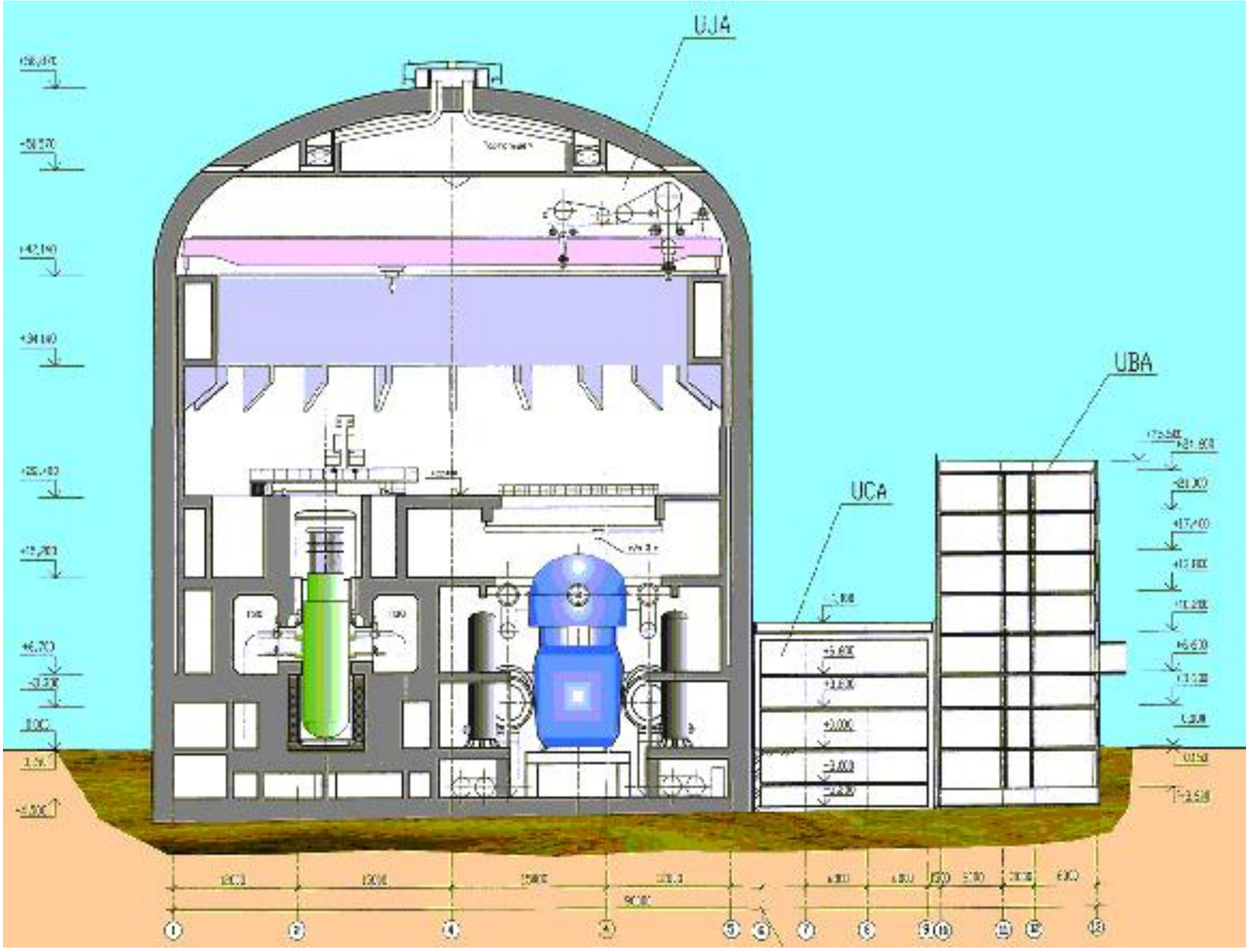
Safety Features

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.

At the same time the power unit design stipulates that the whole power unit will be within a leak-tight enclosure (the secondary containment). The containment accommodates the PC with the VK-300 reactor, emergency cooldown tanks, turbine, spent fuel storage pools, refuelling machine and central hall crane (see next figure). The containment leak rate is 50% of the volume per day with the design pressure of not more than 0.15 MPa. Thanks to new layout concepts for the main equipment of the VK-300 power unit, the containment dimensions do not exceed the dimensions of the VVER-1000 reactor containment.



CNPP unit layout

Description of turbine-generator systems

The turbine-generator systems was developed to produce 250 MW(e) electricity in condensing mode and heat of up to 465 MW(th) within a nuclear cogeneration plant for district heating and for seawater desalination. The VK-300 turbine is mostly based on an elements of WWER-1000 turbine. Heat production systems were designed to supply heat with no radioactivity.

Fuel Characteristics and Fuel Supply Issues

The Hexahedron fuel assembly (FA) is formed by 107 of UO_2 ceramic fuel rods with the enrichment of less than 4% similar to VK-50 - WWER fuel. There are 313 FAs in the core. Fuel burnup is 41.4 GWd/ton.

The integral arrangement of reactor components and availability preliminary and secondary containments are non-proliferation feature of VK-300.

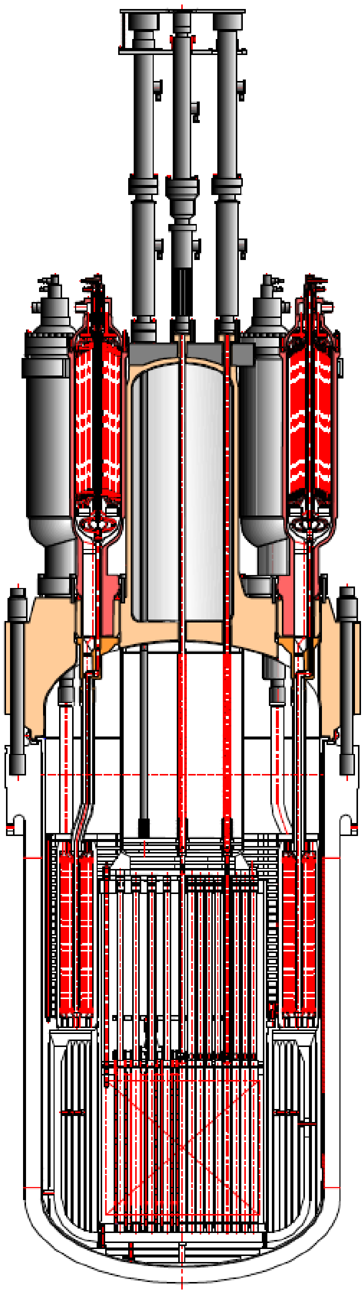
Licensing and Certification Status

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

UNITHERM (RDIPE, Russian Federation)

Introduction

UNITHERM is developed based upon NIKIET's experience in designing marine nuclear installations. The reactor produces 30 MW(th) while the electrical output is rated at 6.6 MW(e). The design assumes that most of the fabrication, assembly, and commissioning of the nuclear power plant modules can be done at a plant. The UNITHERM reactor operates for 20-25 years without refuelling. The land-based siting NPP or barges siting conditions are both viable for the UNITHERM reactor design. NPP with UNITHERM may consists of a number of units depending on purpose and demand.



Reactor System Configuration of UNITHERM

Development Milestones

1994	The NPP design on the base of UNITHERM concept has become the laureate of the competition for SS NPP designs established by the Russian Nuclear Society

Target Applications

The UNITHERM NPP can be used as a source of energy for the generation of electricity, district heating, seawater desalination and process steam production.

In general, configuration and design of the UNITHERM is sufficiently flexible to be adjusted or modified for different target functions and user requirements, without compromising the underlying principles of the concept.

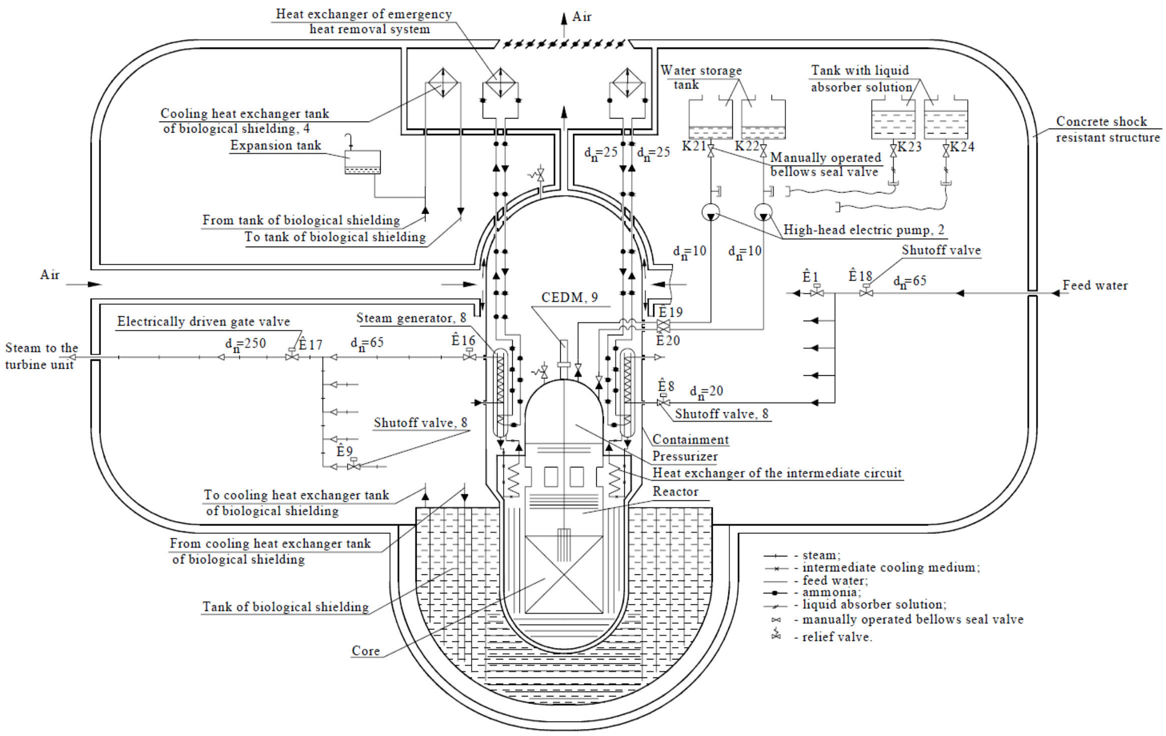
Specific Design Features

One of the main conceptual features of the UNITHERM is the option of shop fabrication and testing.

The integral reactor for land-based option is placed inside the leak-tight containment, which is located within the concrete shock-resistant structure together with the biological shielding and reactor unit components. This structure enhances physical protection of the reactor unit from external impacts such as airplane crash, hurricane, tsunami, unauthorized access, etc.

The UNITHERM plant is designed to operate without operating personnel in attendance. The intent is to provide the reactor maintenance in routine and urgent cases from regional centres common to several plants of this kind.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	Research and Development Institute of Power Engineering, (RDIPE)
Country of Origin:	Russian Federation
Reactor Type:	PWR
Electrical Capacity (MW(e)):	6.6
Thermal Capacity (MW(th)):	30
Expected Capacity Factor (%):	70
Design Life (years):	25
Plant Footprint (m^2):	~ 10000
Coolant/Moderator:	High purity water
Primary Circulation:	Natural circulation
System Pressure (MPa):	16.5
Main Reactivity Control Mechanism:	Soluble boron, rod insertion
RPV Height (m):	9.8
RPV Diameter (m):	2.9
Coolant Temperature, Core Outlet (°C):	330
Coolant Temperature, Core Inlet (°C):	249
Integral Design:	Integral reactor with an integrated pressurizer
Power Conversion Process:	Direct Rankine Cycle
High-Temp Process Heat:	Vapor flow 10.17 kg/c at 285ºC
Low-Temp Process Heat:	Water flow 200 kg/c at 90ºC
Cogeneration Capability:	Heat Mode power: 2.5 MW(e), 20 MW(th) heat
Design Configured for Process Heat Applications:	Use of different type of turbogenerator
Passive Safety Features:	Adjusting elements drive mechanisms with systems blocking the extraction of adjusting elements from reactor core during reactor operation; permanently operational autonomous passive system for abstract of heat from reactor; additional water tank in the upper part of the reactor; guard vessel; containment; iron-water biological protection of reactor; passive systems for abstract of heat from the guard vessel and biological shielding tanks
Active Safety Features:	primary coolant circuit liquid neutrons moderator injection system
Fuel Type/Assembly Array:	UO_2 particles in a metallic (silumin or zirconium) matrix , metal-ceramic/ 54 – 55
Fuel Active Length (m):	1.1
Number of Fuel Assemblies:	265
Fuel Enrichment (%):	19.75
Fuel Burnup (GWd/ton):	1.15
Fuel Cycle (months):	200
Number of Safety Trains:	2
Emergency Safety Systems:	Passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	45 – 50
Distinguishing Features:	Unmanned rector operation
Modules per Plant:	As per customer requirement
Estimated Construction Schedule (months):	84
Seismic Design:	VIII-IX-MSK 64
Predicted Core Damage Frequency (per reactor year):	10^{-6}
Design Status:	Reactor facility and plant conceptual design



UNITHERM schematic diagram (Courtesy of RDIPE, with permission)

Safety Features

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;
- Description of turbine-generator systems.

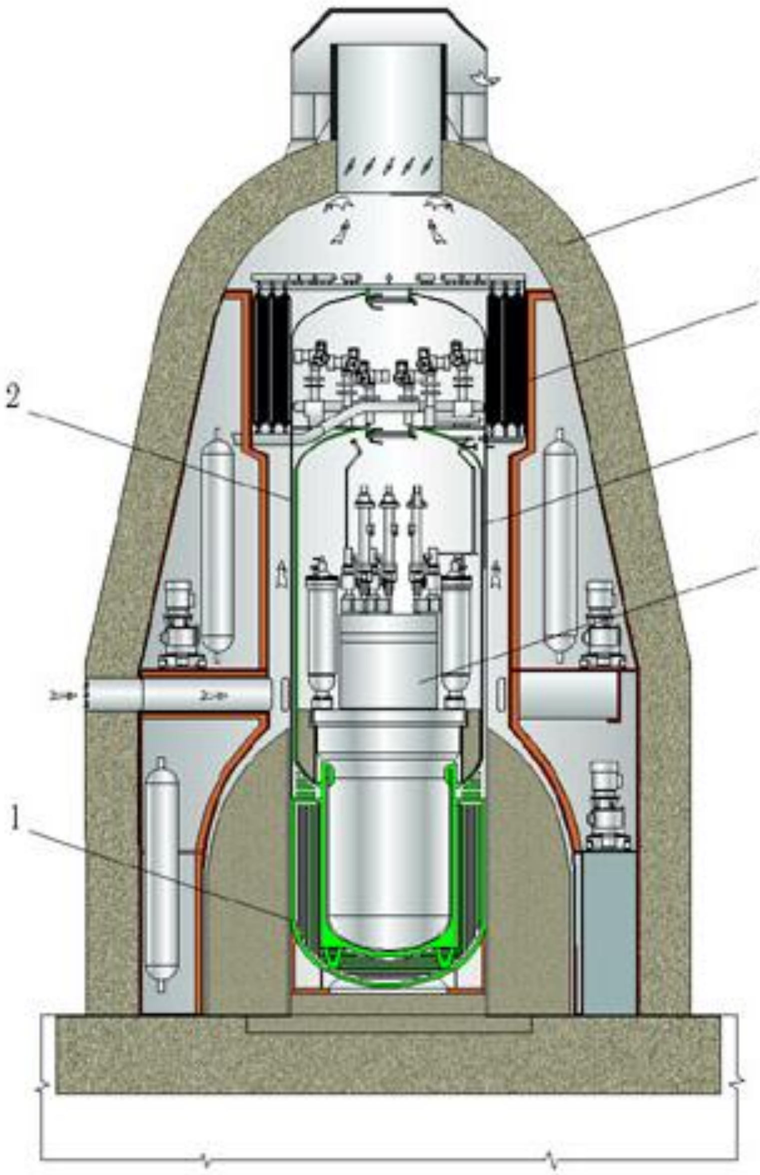
The choice of a candidate turbine-generator plant for the UNITHERM nuclear power plant depends on the plant capacity and operation mode requested by its users.

Description of the Turbine-Generator Systems

Turbine-generator assembly for UNITHERM NPP depends on the plant capacity and operation mode requested by its users.

Fuel Characteristics and Fuel Supply Issues

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.



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

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.

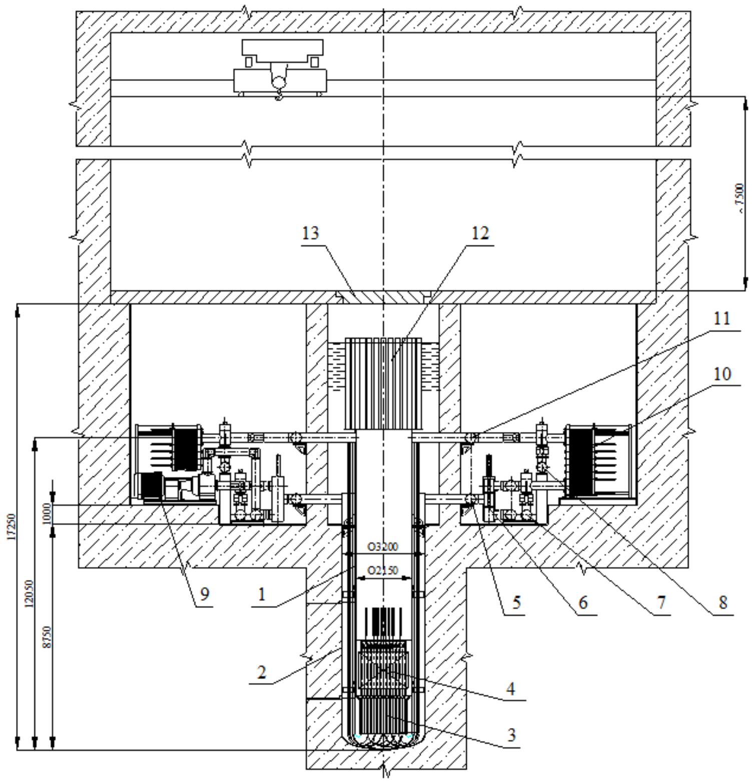
Licensing and Certification Status

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

RUTA-70 (RDIPE, IPPE, Russian Federation)

Introduction

RUTA-70 is 70 MW(th) integral pool-type heating reactor. Reactor pool is made of reinforced concrete lined by stainless steel. The heat from the core to the primary heat exchanger (HX) is transferred by forced convection of the primary water coolant at full power operation but by natural convection under operation conditions below 30% of the rated power. The period of continuous operation of the reactor equipment without a need of maintenance is about one year. Simplicity, high reliability and inherent safety features of the RUTA are based on the low pressure and temperature of the primary coolant as well as integral design of the reactor. Due to high safety features, nuclear district heating plant (NDHP) using RUTA reactors could be located in the immediate vicinity of the heat users.



(1) – riser shroud;
(2) – pool metallic liner;
(3) – core supporting plate with control rod lead tubes;
(4) – reactor core;
(5) – plenum;
(6) – check valve;
(7) – secondary water inlet;
(8) – secondary water outlet;
(9) – primary pump;
(10) – primary HX;
(11) – upper header;
(12) – control rod drives;
(13) – isolation plate.

Reactor System Configuration of RUTA-70
(Courtesy of RDIPE and IPPE, with permission)

Development Milestones

1990	Conceptual design of the 20 MW(th) RUTA heating plant
1992	Feasibility study 'Design and development of the underground NHP with the RUTA reactor for district heating in Apatity-city, Murmansk region'
1994	Feasibility study 'Underground NHP with 4 × 55 MW(th) RUTA reactors for district heating in Apatity-city, Murmansk region'
2003	Technical and economic assessments for using 70 MW(th) RUTA reactor to upgrade the district heating system in Obninsk, Kaluga region. Feasibility study of construction a NDHP in the State Research Centre of the Russian Federation - IPPE, Obninsk. Approval of the project by the Board for the programme 'Development of Obninsk as a 'science' town'

Target Applications

The RUTA concept and design is primarily developed to provide district heating in remotely isolated areas of Russia suffering lack of fossil fuel. Continuous increase of the organic fuel costs in the country essentially broaden the area of competitive application of RUTA as a heating reactor. In addition, promising way of commercial application of low potential thermal energy generated by the RUTA reactor is distillation process for seawater and brackish water desalination.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	RDIPE and IPPE
Country of Origin:	Russian Federation
Reactor Type:	Pool-type
Electrical Capacity (MW(e)):	N/A
Thermal Capacity (MW(th)):	70
Expected Capacity Factor (%):	95
Design Life (years):	60
Plant Footprint (m^2):	100000
Coolant/Moderator:	Water
Primary Circulation:	Natural (below 30% of N)/Forced (for 30-100% of N)
System Pressure:	Atmospheric pressure at reactor poll water surface
Main Reactivity Control Mechanism:	Control Rod Driving Mechanisms, Burnable poison
RPV Height (m):	N/A, Reactor Pool Height - 17.250 m
RPV Diameter (m):	N/A, Lower part of Reactor Pool Diameter - 3.2 m
Primary Coolant Temperature, Core Outlet (°C):	101
Primary Coolant Temperature, Core Inlet (°C):	75
Integral Design:	Yes
Power Conversion Process:	N/A
High-Temp Process Heat:	N/A
Low-Temp Process Heat:	Yes
Cogeneration Capability:	N/A
Design Configured for Process Heat Applications:	Yes
Passive Safety Features:	Yes
Active Safety Features:	Yes
Fuel Type/Assembly Array:	Cermet (0.6 UO_2+ 0.4 Al alloy)/hexagonal
Fuel Active Length (m):	1.40
Number of Fuel Assemblies:	91
Fuel Enrichment (%):	3
Fuel Burnup (GWd/ton):	25-30
Fuel Cycle (months):	36
Number of Safety Trains:	2
Emergency Safety Systems:	Active and Passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	28
Distinguishing Features:	Coupling with desalination system, isotope production, medical neutron beams
Modules per Plant:	1
Estimated Construction Schedule (months):	36
Seismic Design (g):	>0.8 (automatic shutdown)
Predicted Core Damage Frequency (per reactor year):	PSA is not completed, assessment < 10^{-6}
Design Status:	Conceptual design

Specific Design Features

The reactor core is located in the lower part of the reactor pool, in the shell of the chimney section. The distributing header is placed in the upper part of the shell of the chimney section. Heat exchangers are located in a compartment in the upper part of the pool. The HXs can be passively flooded by water from the pool providing long term heat sink in case of emergency.

Pumps and HX are located as such to allow easy access for inspection or replacement. Two axial pumps are installed in the primary circuit.

The protective flooring composed of slabs that are installed above the reactor pool to avoid possible damage to the primary components from external impacts. To prevent gas and vapour penetration to the reactor hall from the upper part of the reactor, joints of the protective slabs are gas-tight.

Safety Features

The safety concept of the RUTA-70 is based on optimum use of inherent and passive safety features to secure a high level of the reactor safety. The acceptable safety level is understood as the level when the total effective annual internal and external exposure radiation dose for the population under normal operating and emergency conditions does not exceed the natural background dose.

The distinctive feature of pool type reactors is that there is no excess pressure in the reactor pool; this excludes an accident with an instantaneous rupture of the primary circuit and cessation of heat transfer from the core due to dry out.

The high heat accumulating capability of water in the reactor pool ensures slow changing of coolant parameters during transient and emergency conditions and reliable heat transfer from the fuel, even if controlled heat transfer from the reactor is not available. Fuel temperatures are moderate.

The RUTA-70 uses mostly passive systems to perform safety functions such as: air heat sink system for emergency cooldown (ASEC), gravity driven insertion of the control rods in the core as reactor safety control system, the secondary circuit overpressure protection system, the overpressure protection system for air space in the reactor pool and prestressed concrete external impacts protection system.

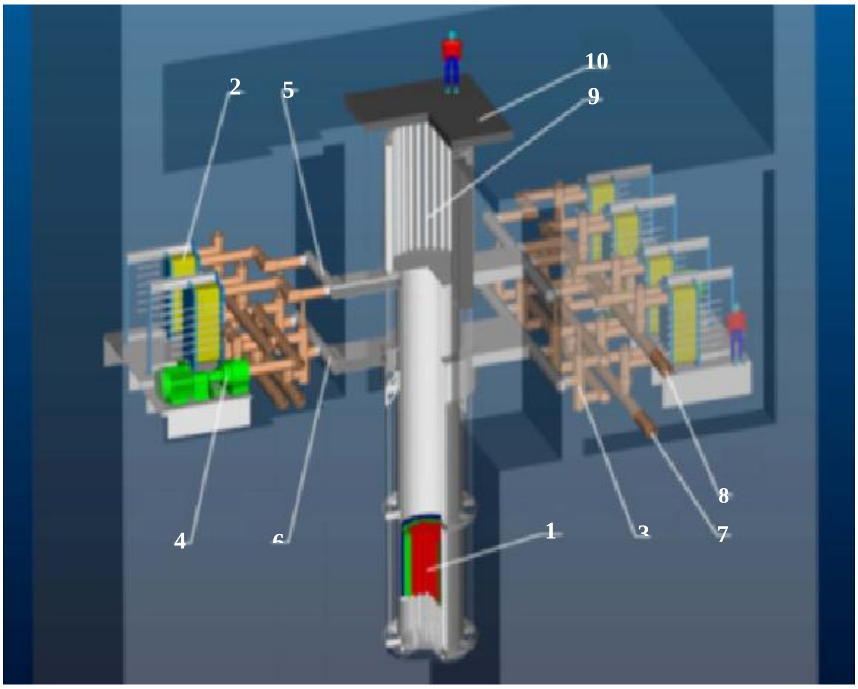
The computational analysis of beyond design basis accidents has revealed that in case of multiple failures in the reactivity control systems and devices, safety can be ensured by self-control of reactor power (boiling - self-limitation of power), i.e. through the inherent safety features of the reactor.

If all controlled trains of heat removal are lost, heat losses via the external surface of the reactor pool to the surrounding environment (ground) are considered as an additional safety train. Residual heat is accumulated in the pool water. The transient of pool water heatup in the aqueous mode before the onset of boiling takes several days. As soon as boiling starts, steam goes to the reactor hall where it is condensed (passive condensing facilities are provided). A reactor boil-off without makeup takes 18 to 20 days. Upon completion of this period residual heat is balanced by heat transfer to the ground. Core dry out is avoided. Moderate temperatures are not exceeding the design limits characterize fuel elements.

An unrecoverable leak in the reactor pool will not occur due to the concrete pool which is designed to withstand external events, including maximum design basis earthquake and water filtration to the ground in a beyond design basis accident.

Fuel Characteristics and Fuel Supply Issues

According to the design of the NHP RUTA-70, spent fuel assemblies should be stored in the cooling pond for 3 years after discharge from the reactor core and then transported to the fuel reprocessing plant without further long-term on-site storage.



RUTA-70 Plant layout (Courtesy of RDIPE and IPPE, with permission)
(1) – Core; (2) – Primary heat exchanger; (3) – Check valve; (4) – Pump;
(5) – Primary circuit distributing header; (6) – Primary circuit collecting header;
(7) – Secondary circuit inlet pipeline; (8) – Secondary circuit outlet pipeline;
(9) – SCS drives; (10) – Upper slab

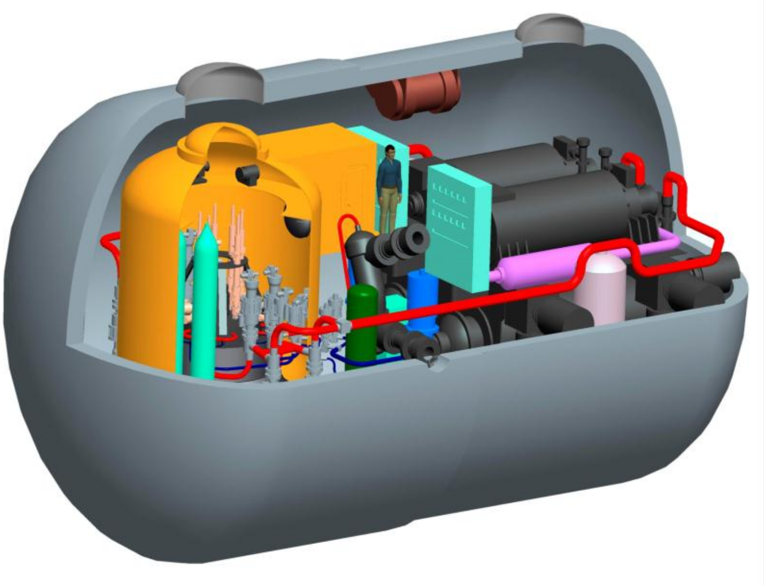
Licensing and Certification Status

To provide an operating reference for the reactor, the Institute of Physics and Power Engineering (IPPE) proposed to build the first reference RUTA-70 unit in Obninsk, where the institute is located. The proposal, developed in 2004, was supported by the feasibility study jointly carried out by NIKIET, IPPE, and Atomenergoproekt (Moscow). This study showed that RUTA-70 should be ultimately deployed along with the non-nuclear sources of power operating in peak and off-peak mode. For the current status, RUTA is still in the conceptual design stages.

SHELF (RDIPE, Russian Federation)

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.



Reactor System Configuration of SHELF (Courtesy of RDIPE, with permission)

Specific Design Features

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.

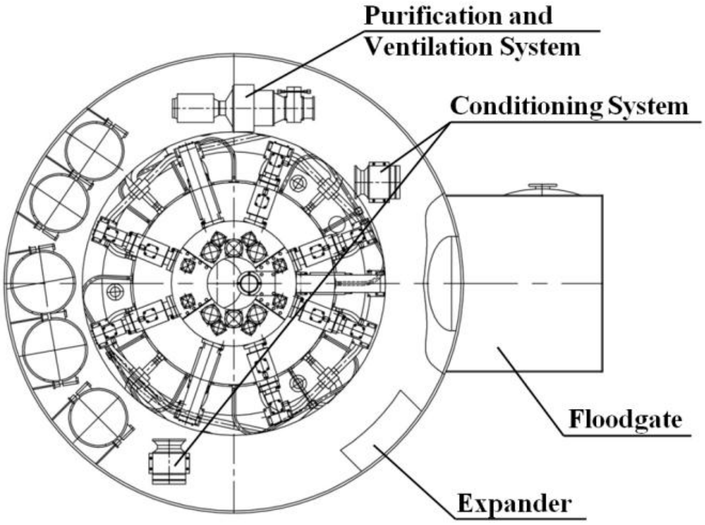
Safety Features

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 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.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	Research and Development Institute of Power Engineering, (RDIPE)
Country of Origin:	Russian Federation
Reactor Type:	PWR
Electrical Capacity (MW(e)):	6.0
Thermal Capacity (MW(th)):	28
Design Life (years):	30
Coolant/Moderator:	Light water
Primary Circulation:	Forced and natural circulation
System Pressure (MPa):	17
Main Reactivity Control Mechanism:	Rod insertion
Coolant Temperature, Core Outlet (°C):	320
Integral Design:	Yes
Power Conversion Process:	Direct Rankine Cycle
Fuel Type:	UO_2 and aluminium alloy matrix
Fuel Enrichment (%):	< 20
Fuel Cycle (months):	56
Number of Safety Trains:	2
Emergency Safety Systems:	Active
Residual Heat Removal Systems:	Passive
Distinguishing Features:	Underwater energy source
Seismic Design:	7.0g
Predicted Core Damage Frequency (per reactor year):	10^{-6}
Design Status:	Conceptual design

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.



SHELF containment top view

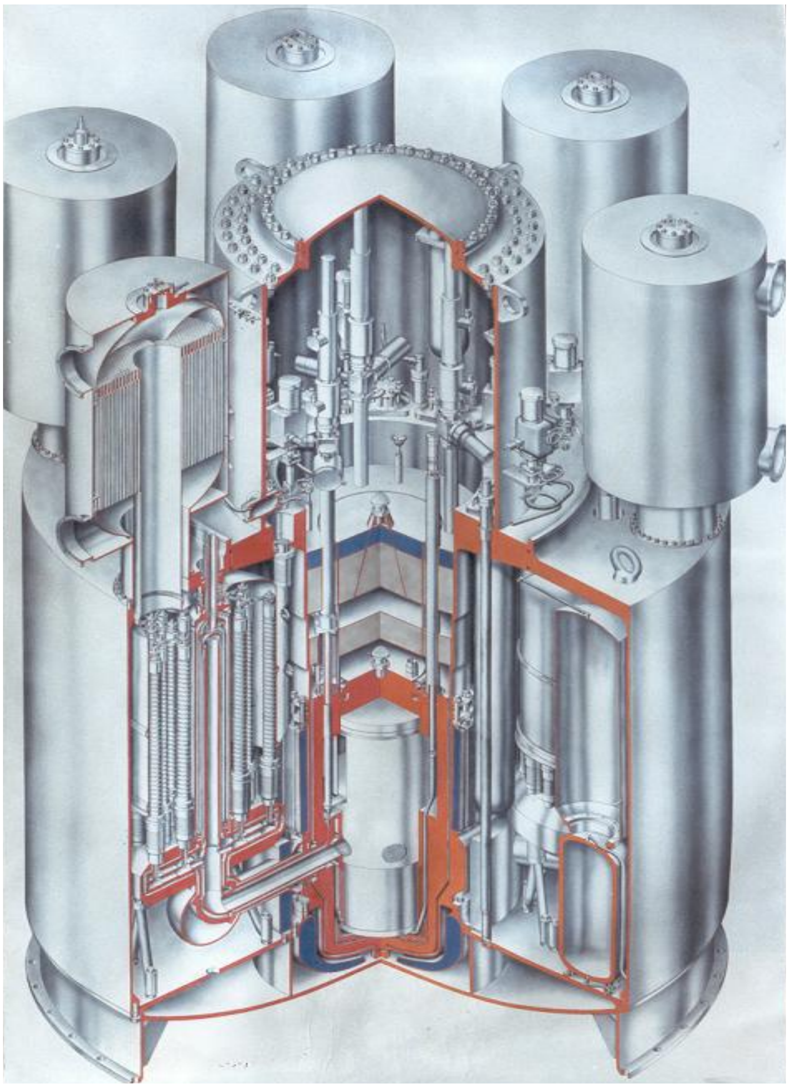
Licensing and Certification Status

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

ELENA (Research Russian Centre "Kurchatov Institute", Russian Federation)

Introduction

ELENA is a direct conversion water-cooled reactor capable to supply electricity and heat over a 25 year life of the plant without refuelling. This is a very small reactor with just 68 kW(e) in power generating capacity and another 3.3 MW(th) of heating capacity. The key aspect of this design is that it is meant to be an "unattended" nuclear power plant, requiring nearly no operating or maintenance personnel over the lifetime of the unit. The ELENA NTEP project was developed using the experiences in construction and operation of marine and space power plants and the operation experience of the GAMMA reactor. The concept has been developed by the Russian Research Centre "Kurchatov Institute" (RRC KI). The ELENA NTEP is a land-based plant; however, in principle it is possible to develop versions for underground or underwater deployment. The reactor and its main systems are assembled from factory-fabricated finished units, whose weight and dimensions enable any transport delivery for the complete plant, including helicopter and ship.



Reactor System Configuration of ELENA

Target Applications

The unattended ELENA NTEP plant is designed to produce heat for towns with a population of 1500–2000 located in remote areas where district heating is required. Since it is auxiliary in nature, the electricity generation of 68 kW could be used for the in-house power needs of the plant and to supply electricity to consumers requiring a highly reliable power supply, such as hospitals, etc.

A desalination unit can be used in combination with the ELENA NTEP plant.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	Russian Research Centre “Kurchatov Institute” (RRC KI)
Country of Origin:	Russian Federation
Reactor Type:	PWR
Electrical Capacity (MW(e)):	0.068
Thermal Capacity (MW(th)):	3.3
Coolant/Moderator:	Light water
Primary Circulation:	Natural circulation
System Pressure (MPa):	19.6
Power Conversion Process:	Heating reactor, direct, thermo-electric
Coolant Temperature, Core Outlet (°C):	328
Coolant Temperature, Core Inlet (°C):	311
Fuel Type/Assembly Array:	UO_2 pellets; MOX fuel is an option
Number of Fuel Assemblies:	109
Fuel Enrichment (%):	15.2
Fuel Burnup (MW day/t U):	57600/27390
Fuel Cycle (months):	300
Design Status:	Conceptual design

Specific Design Features

The specific features of the design are:

- Capability of power operation without personnel involvement;
- Compensation of burn-up reactivity swing and other external reactivity disturbances without moving the control rods;
- The use of thermoelectric energy conversion to produce electricity.

The system for heat transport from the core to the consumer has four circuits:

- Circuit II (intermediate circuit) removes heat from the "cold" joints of the thermal elements and transfers it through natural circulation to the intermediate heat exchanger of circuits II–III; the coolant is specially treated water, which also acts as part of the steel-water radiation shielding;
- Circuit III transfers heat through natural circulation to the heat exchanger of the heat supply circuit; the coolant is ethanol;
- Circuit IV transfers heat from the heat exchanger of circuits III–IV to the consumers using forced circulation; the circuit IV coolant is A-60 antifreeze.

The reactor is designed to operate in a base load mode. A decrease in the heat or consumer power is automatically compensated through the discharge of excess heat to the atmosphere via a dry cooling tower, with no changes in the electric power.

The reactor is installed in a caisson forming a heat-insulating gas cavity in the area of the strengthened reactor vessel and a caisson space above the reactor cover to house control and protection system (CPS) drives and to prevent radioactive substances from escaping into the surrounding space in case of a circuit I break.

The reactor installation is based on passive principles of heat removal (natural convection in all circuits, except for heat transport to the consumers) in normal operation and in shutdown conditions.

The active components of the protection system are scram actuators for six compensation groups of the control rods.

The control safety system (CSS) consists of a control safety system for emergency shutdown and a system to input, process and transmit safety-related plant information. During normal operation the emergency shutdown CSS is permanently awaiting a scram actuation request; it also periodically provides information on the state of the plant.

The ELENA-NTEP CSS has three independent power supplies: 2 TEG sections, a diesel generator, and a storage battery.

The localizing safety systems provide the defence in depth and secure the plant safety based on inherent safety features and predominantly passive phenomena; they require no human intervention or external power sources.

The safety barriers of the ELENA-NTEP are:

- The fuel elements;
- The leak-tight primary circuit;
- The caisson;
- The reactor vessel and the guard vessel designed to withstand the pressure arising within each of them at their consecutive failure; and
- An embedded silo sealed with the protective plate.

Special measures for the protection of hot water consumers ensure that radioactivity is never released into the network circuit.

Safety support systems create the conditions required for normal functioning of the safety systems; they include power supply systems and a heat removal system that transmits heat to the consumers.

Description of the Thermoelectric Generator

A thermoelectric generator (TEG) is used as a heat exchanger between circuits I and II; it is based on semiconductor thermo-elements enabling the generation of 68 kW of power in the reactor nominal operating mode simultaneously with heat transfer to circuit II. This power is used for the plant auxiliary needs; it could also be supplied to a small town without district power supply, partially replacing a diesel power plant.

The TEG consists of eight identical thermoelectric units (TEU). Each of them includes 36 thermoelectric modules equipped with thermoelectric packs of bismuth tellurides with electronic and hole conduction.

Fuel Characteristics and Fuel Supply Issues

Pellet type uranium dioxide fuel is used with the average ^{235}U enrichment of 15.2%; the neutron moderator and coolant is water specially treated according to specified water chemistry. Cylindrical fuel elements with stainless steel cladding are installed in 109 fuel assemblies of 55 fuel elements each; 216 absorber rods with boron carbide based neutron absorber are divided into 6 groups. Fuel assemblies also include burnable absorbers made of Gd-Nb-Zr alloy. The ^{235}U load is 147 kg.

The cylindrical core with a height of 850 mm and the equivalent diameter of 833 mm is installed in a steel shell with a diameter of 920 mm and is encircled by an iron-water shield. The strengthened stainless steel reactor vessel has an internal diameter of 1250 mm and a height of 3700 mm with a wall thickness of 132 mm.

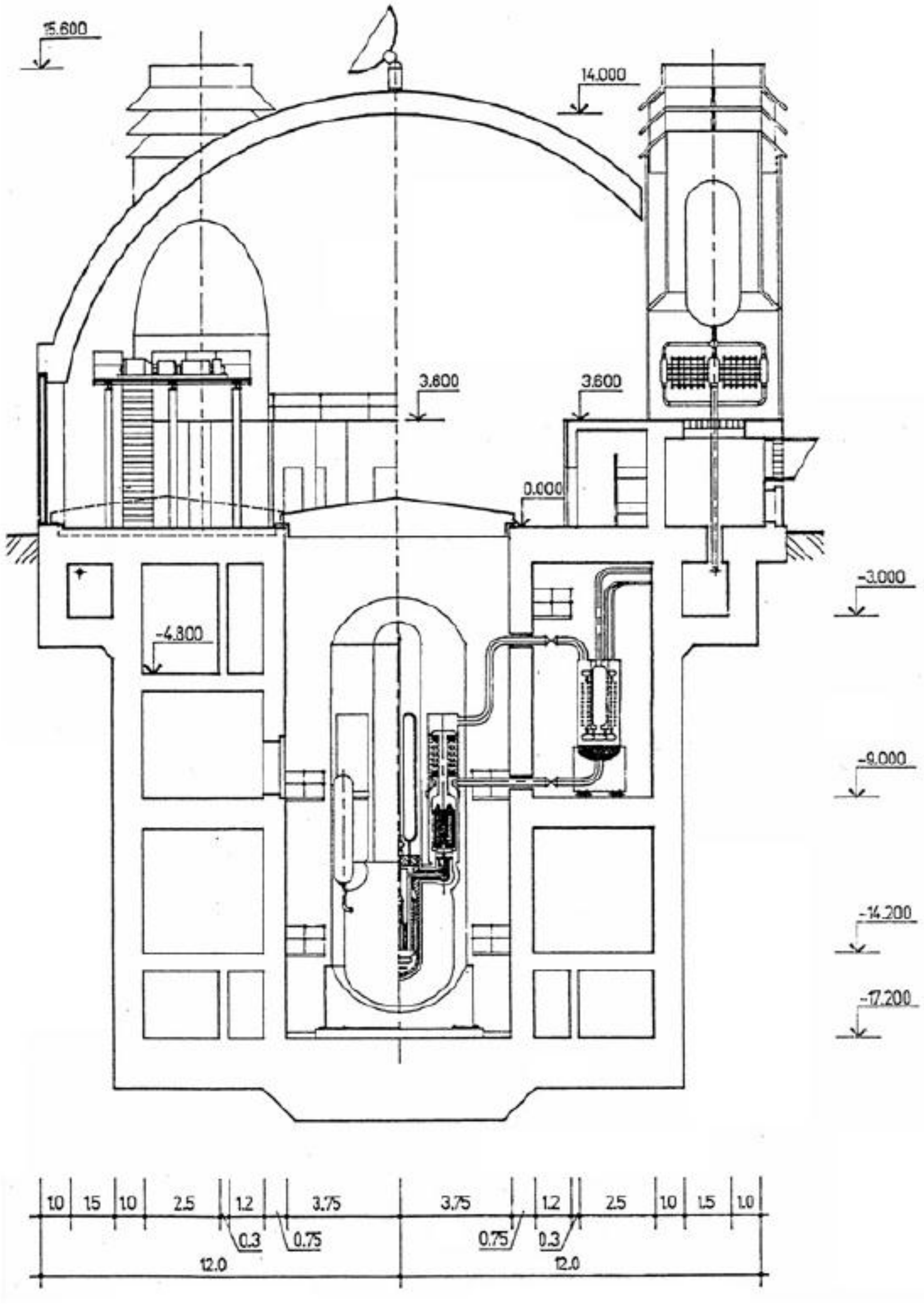
Plant Layout

The plant includes instrumentation and control systems; a system for heat removal to consumers; an auxiliary power supply system; and a radiation monitoring system, including process radiation monitoring, dosimetric monitoring, and environmental monitoring.

The plant has a main control and monitoring room accommodating the start-up and instrumentation and control equipment, as well as the equipment necessary to prepare the information to be transmitted to a monitoring centre.

The plant building has a cylindrical shape and is embedded in the ground for the entire reactor installation height with a foundation plate elevation of –19.2 m. The elevation of +0.0 has a domed ceiling. The underground portion of the structure, the walls and the overlaps are monolithic reinforced concrete.

The plant incorporates a physical protection system, has a fence and is equipped with external lighting.

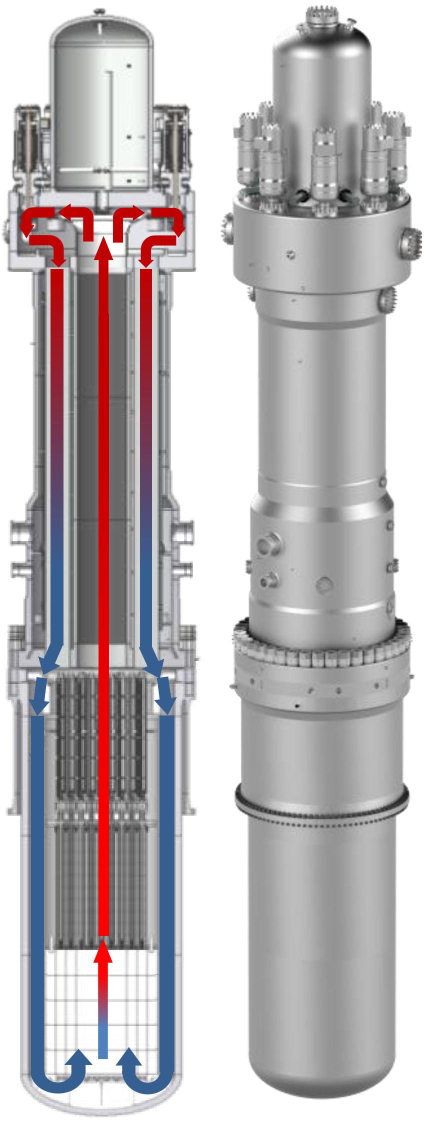


Plant general view of ELENA

mPower (B&W Generation mPower, USA)

Introduction

The mPower™ small modular reactor is an integral PWR, designed by Generation mPower and its affiliates Babcock & Wilcox mPower, Inc. and Bechtel Power Corporation, to generate a nominal output of 180 MW(e) per module. In its standard plant design, each mPower Plant is comprised of a 'twin-pack' set, or two mPower units, generating a nominal 360 MW(e). The design adopts internal steam supply system components, once-through steam generators, pressurizer, in-vessel control rod drive mechanisms (CRDMs), and horizontally mounted canned motor pumps for its primary cooling circuit and passive safety systems. The facility structure is composed of reactor modules that are fully shop-manufactured, rail-shippable to a site and installed into the facility, with the capability to install modules on an as-needed basis to meet demand growth. The plant is designed to minimize emergency planning zone requirements.



Reactor System Configuration of mPower (Courtesy of Generation mPower, with permission)

Development Milestones

2009	B&W officially introduced the mPower SMR
2012	The Integrated System Test (IST) facility located in Bedford County, Virginia, was put into operation

Target Applications

The primary application for the mPower reactor is electricity production. The mPower design could be retrofitted to support other heat-requiring or cogeneration applications.

Specific Design Features

Currently, 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 reactor is electrically heated and the coolant pressure is nominally 14.2 MPa. The steam generator (SG) assemblies are located within the annular space formed by the inner reactor pressure vessel (RPV) walls and the riser surrounding and extending upward from the core. The SG assembly may be removed at once as it is isolated from the RPV bottom section by a sealing flange, so that the top RPV sections covering the SGs can be disconnected from the core-bearing RPV bottom portion and repositioned for refuelling or maintenance purposes. The CRDM design is fully submerged in the primary coolant within the RPV boundary excluding the possibility of control rod ejections accident scenarios. Reactivity control of mPower design is achieved through the electro-mechanical actuation of control rods only (i.e., no soluble boron). The NSSS forging diameter allows for greater sourcing options and rail shipment.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	B&W Generation mPower (B&W)
Country of Origin:	USA
Reactor Type:	Integral PWR
Electrical Capacity (MW(e)):	180
Thermal Capacity (MW(th)):	530
Expected Capacity Factor (%):	95
Design Life (years):	60
Plant Footprint (m^2):	160 000
Coolant/Moderator:	Light water
Primary Circulation:	Forced circulation
System Pressure (MPa):	14.2
Main Reactivity Control Mechanism:	Control rods
RPV Height (m):	27
RPV Diameter (m):	4
Coolant Temperature, Core Outlet (°C):	318.8
Coolant Temperature, Core Inlet (°C):	295
Integral Design:	Yes
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	No
Low-Temp Process Heat:	Possible
Cogeneration Capability:	Possible
Design Configured for Process Heat Applications:	No
Passive Safety Features:	Yes
Active Safety Features:	Yes
Fuel Type/Assembly Array:	UO_2 pallet/17x17 square
Fuel Active Length (m):	2.4
Number of Fuel Assemblies:	69
Fuel Enrichment (%):	< 5.0
Fuel Burnup (GWd/ton):	> 40
Fuel Cycle (months):	48
Number of Safety Trains:	2
Emergency Safety Systems:	Passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	15
Distinguishing Features:	Internal once-through steam generator, pressurizer and control rod drives
Modules per Plant:	2
Estimated Construction Schedule (months):	36
Seismic Design:	Target 85% of contiguous US
Predicted Core Damage Frequency (per reactor year):	Target 10^{-8}
Design Status:	Basic design

Safety Features

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 any safety credit for standby diesel generators, and a design objective is no core uncovery occurs during any credible design basis accident. 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 SMR has decay heat removal systems that consist of a passive heat exchanger connected with the atmosphere (as 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. The below grade installation of the entire NSSS and the integral RPV within the containment vessel enhance the ability to cope with seismic events. The safety features aim for long term coping time without off-site power.

Electrical, and Instrumentation and Control Systems

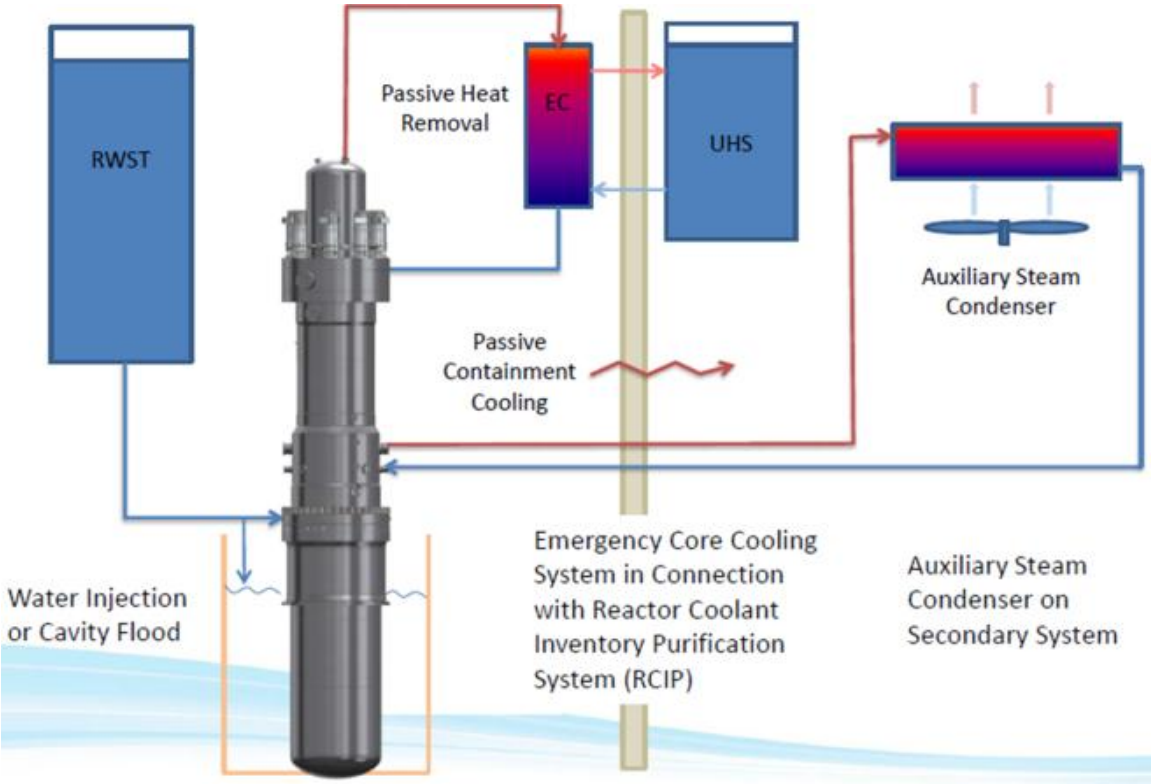
Digital instrumentation and control systems (I&C) are adopted for the mPower SMR. The digital I&C enables high level of plant automation, including control of startup, shutdown and load following. The digital control system architecture has also been developed. As a result of the mPower SMR I&C design and digital control system architecture, the USNRC I&C staff took this opportunity to restructure and revise the current guidance contained in Chapter 7, “Instrumentation and Controls,” of the Standard Review Plan (SRP) for its review of the mPower SMR.

Description of Turbine-Generator Systems

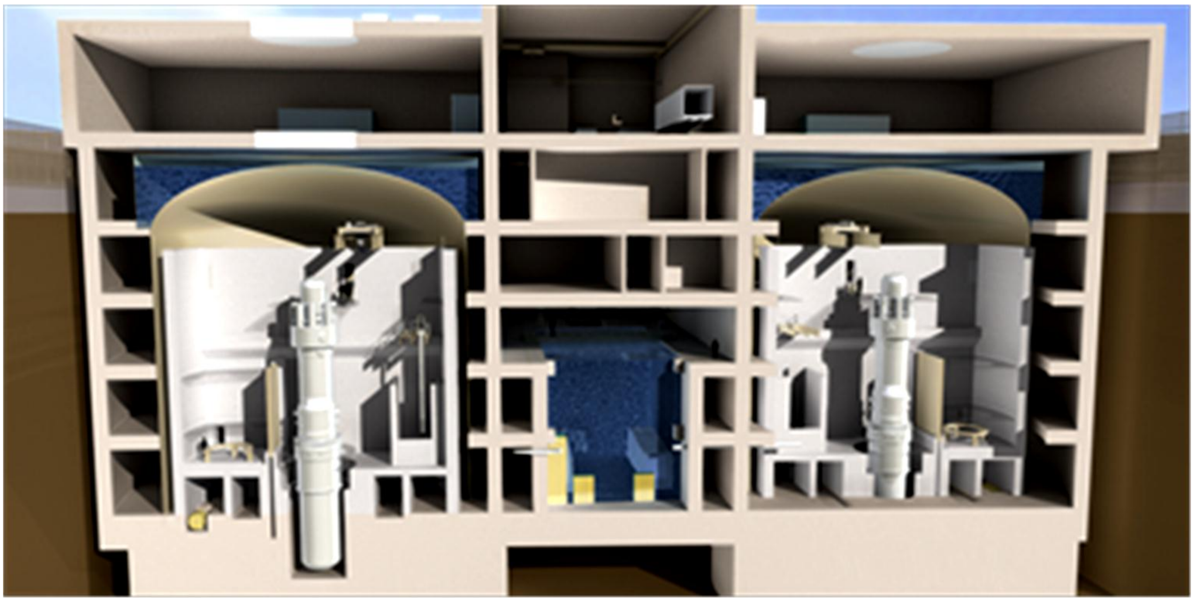
The balance of plant design (BOP) consists of a conventional power train using a steam cycle and a water-cooled (or optional air-cooled) condenser. While site conditions and specifics of the turbine/generator and condenser determine the final power output of the plant, the water cooled condenser provides for a nominal output power of 180 MW(e), while an air cooled condenser would nominally provide 155 MW(e). The conventional steam cycle equipment is small, easy to maintain and replace. The BOP operation is not credited for design basis accidents. The feedwater system of mPower accommodates air-cooled or water-cooled condensers. The generator is designed for power manoeuvring and flexible grid interface.

Fuel Characteristics and Fuel Supply Issues

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 B4C 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. As mPower has the longer 48-month fuel cycle, 100% of the steam generator tubes will be inspected during each outage.



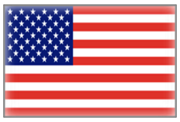
Decay heat removal strategy (Courtesy of Generation mPower, with permission)



mPower containment design (Courtesy of Generation mPower, with permission)

Licensing and Certification Status

In 2011, affiliates of 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 four B&W mPower reactors. B&W submitted an application to the US Department of Energy for the SMR development support programme and received the first DOE funding for SMR design certification review with the US Nuclear Regulatory Commission. Generation mPower is presently working with B&W and Bechtel to prepare a design certification application for submittal to the NRC.



NuScale (NuScale Power Inc., USA)

Introduction

NuScale reactor is made up of one to twelve independent reactor modules each producing a net electric power of greater than 45 MW(e) resulting in a plant output of greater than 540 MW(e) for a twelve-module plant. The reactor operates based on natural convection instead of using pumps to circulate water through reactor core and adopts fully passive safety features. The NuScale value proposition involves innovative design principles to achieve significant improvement in safety, reduced capital at risk, and flexibility/scalability in plant size and application. Each reactor module includes a high pressure containment vessel immersed underwater in a below-grade pool. The NuScale primary system and containment are prefabricated and transported by rail, truck or barge to the plant site, which shortens construction schedule to approximately 36 months. The integral modular design of NuScale allows new modules to be added to the plant or refuelled independently while the other modules continue to operate.

Reactor System Configuration of NuScale
(Courtesy of NuScale Power Inc., with permission)

Development Milestones

2003	First Integral test facility operational
2007	NuScale Power was formed
2008	US NRC design certification pre-application started
2012	Twelve-reactor simulated control room was commissioned
2016 (2^{nd} half)	NuScale to submit design certification application to the NRC
2020	NuScale design certification target
2023	NuScale commercial operation target

Target Applications

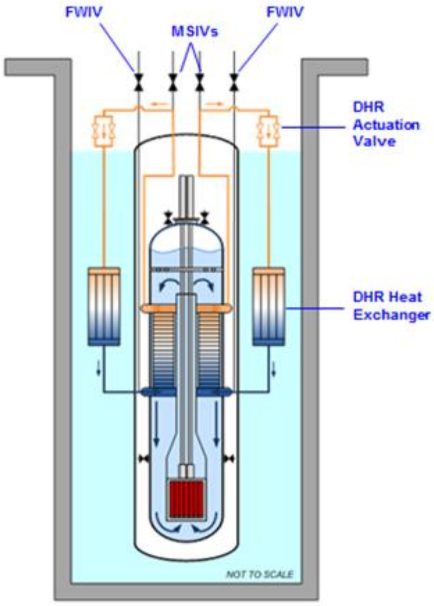
NuScale design is a modular reactor for electricity production and non-electrical process heat applications.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	NuScale Power, LLC
Country of Origin:	USA
Reactor Type:	Integral PWR
Electrical Capacity (MW(e)):	> 45 (net)
Thermal Capacity (MW(th)):	160
Expected Capacity Factor (%):	> 95
Design Life (years):	60
Plant Footprint (m^2):	130 000
Coolant/Moderator:	Light water
Primary Circulation:	Natural circulation
System Pressure (MPa):	12.8
Main Reactivity Control Mechanism:	CRDM, boron
RPV Height (m):	17.4
RPV Diameter (m):	2.9
Integral Design:	Yes
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	No
Low-Temp Process Heat:	Possible
Cogeneration Capability:	Possible
Design Configured for Process Heat Applications:	No
Passive Safety Features:	Yes
Active Safety Features:	No
Fuel Type/Assembly Array:	UO_2 pellet/17x17 square
Fuel Active Length (m):	2
Number of Fuel Assemblies:	37
Fuel Enrichment (%):	< 4.95
Fuel Burnup (GWd/ton):	TBD
Fuel Cycle (months):	24
Number of Safety Trains:	2
Emergency Safety Systems:	Passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	10
Distinguishing Features:	Unlimited coping time for core cooling without AC or DC power, water addition, or operator action as demonstrated
Modules per Plant:	1 – 12
Estimated Construction Schedule (months):	~36
Seismic Design (g):	0.5 peak ground acceleration
Predicted Core Damage Frequency (per reactor year):	10^{-8} (internal events)
Design Status:	Basic design

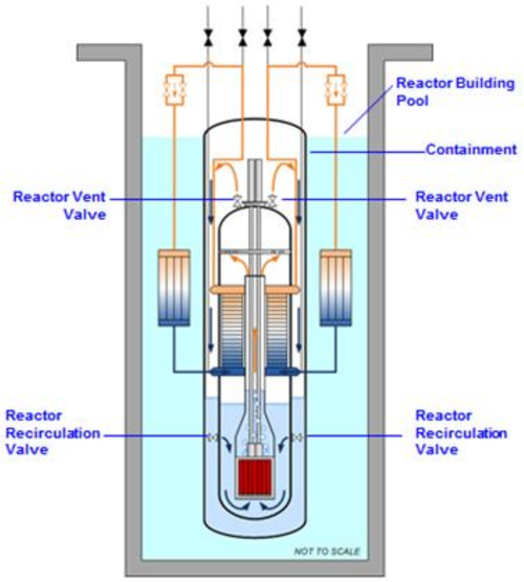
Specific Design Features

The main distinctive feature of this design is that the reactor pressure vessel (RPV) is placed within an additional pressure vessel made of high-strength stainless steel, which is submerged underwater in a below-grade pool shared by all modules. This configuration provides post-accident passive containment cooling and decay heat removal for an unlimited period of time. Heat loss is minimized by maintaining a vacuum between the containment vessel and the RPV to provide thermal insulation.

The NuScale SMR utilizes an integral reactor vessel design with the nuclear core, steam generator, and pressurizer integral within a single vessel. The approximate dimensions are 17.4 m (45 feet) in length and 2.9 m (9.6 feet) in diameter. The NuScale reactor incorporates two independent helical-coil steam generators that are located in annulus between the upper riser and the RPV shell. The feedwater enters the feed plenums, flows upward through the inside of the tubes and is discharged via the steam headers. The reactor coolant flows upward through the upper riser, and then turned by the pressurizer baffle plate and flows down over the shell of the helical tube bundle.



Decay Heat Removal System (NuScale)



Emergency Core Cooling System (NuScale)

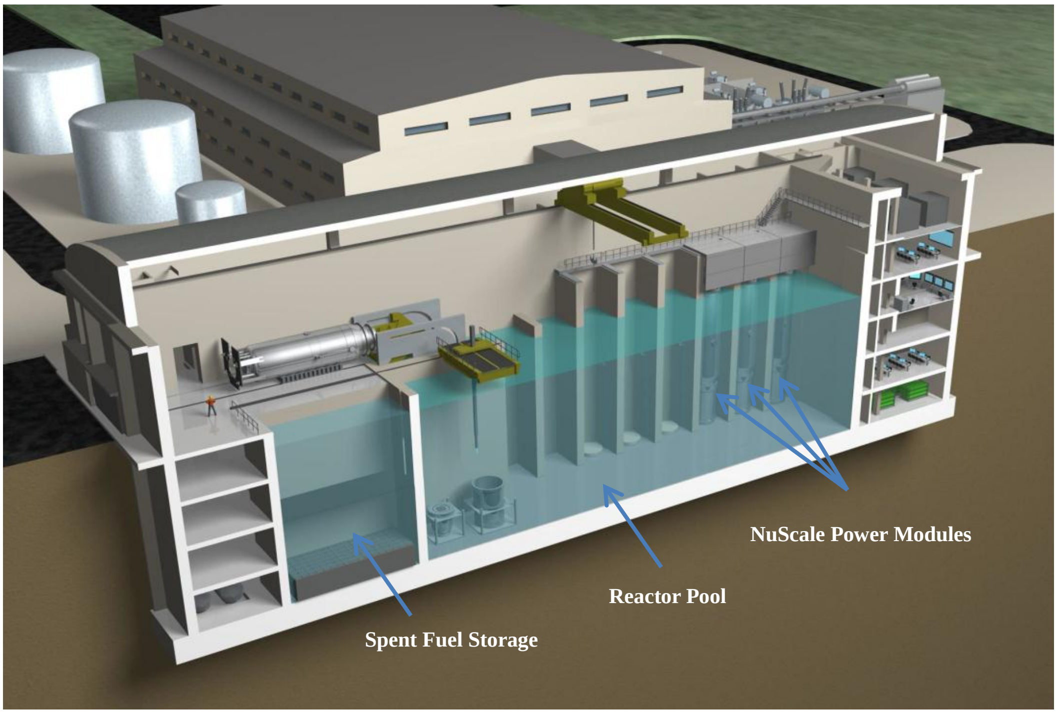
Safety Features

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 systems and an emergency core cooling system. The decay heat removal system consists of two independent trains operating under two-phase natural circulation in a closed loop. The pool surrounding the reactor module provides at least 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 condensation on the containment, condensate collecting in the lower containment region, and reactor recirculation valves to provide recirculation through the core. This is said to provide greater than 30 days of water cooling followed by an unlimited duration of air cooling.

Immersion of the high-strength containment vessel in the reactor pool provides assured access to ultimate heat sink for long-term cooling. The NuScale high-pressure containment paradigm achieves equilibrium pressure between reactor and containment following any loss of coolant accident (LOCA) that is always below containment design pressure. The insulating vacuum space significantly reduces heat transfer during nominal operation and enhances steam condensation inside the containment during reactor vessel blow-down.

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. Each module is self-contained with the individual shutdown protections/systems in case of emergency.

Each NuScale module operates with an independent steam Rankine cycle power conversion unit. A module is disconnected and moved to a disassembly/reassembly location for refuelling and inspections while the remaining modules continue to operate. This approach minimizes grid disruption during outages. Because of the significantly lower core damage frequency afforded by the simplified design features and the reduced accident source term resulting from small unit size and additional radionuclide barriers. The NuScale Emergency Planning Zone (EPZ) is expected to be as near as the site boundary rather than the current 10 miles required by large traditional plants in the U.S.



NuScale plant layout (Courtesy of NuScale Power Inc., with permission)

Electrical, and instrumentation and control systems

The current NuScale design proposes using digital controls for and operating all plant modules from a single control room. 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 the operations staff.

Description of the turbine-generator systems

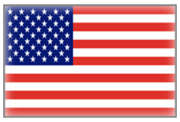
There are individual turbines for each of the reactor modules. They will be skid mounted and standard models are currently available.

Fuel Characteristics and Fuel Supply Issues

Traditional 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 refuelling cycle of 24 months, which is driven by inspection requirements of the U.S. Nuclear Regulatory Commission.

Licensing and Certification Status

The NuScale Integral System test facility is being used to evaluate design performance and improvements, and to conduct integral system tests for NRC certification. NuScale has a target commercial operation date of 2023 for the first plant that is to be built in Idaho.



Westinghouse SMR (Westinghouse Electric Company LLC, USA)

Introduction

The Westinghouse SMR is an integral pressurized water reactor (PWR) that improves on the concepts of simplicity and advanced passive safety demonstrated in the AP1000 plant. The power station delivers a thermal output of 800 MW(th) and an electric output of greater than 225 MW(e) as a stand-alone unit, completely self-contained within a compact plant site. The entire plant is designed for modular construction with all components shippable by rail, truck, or barge.

Reactor System Configuration of Westinghouse SMR

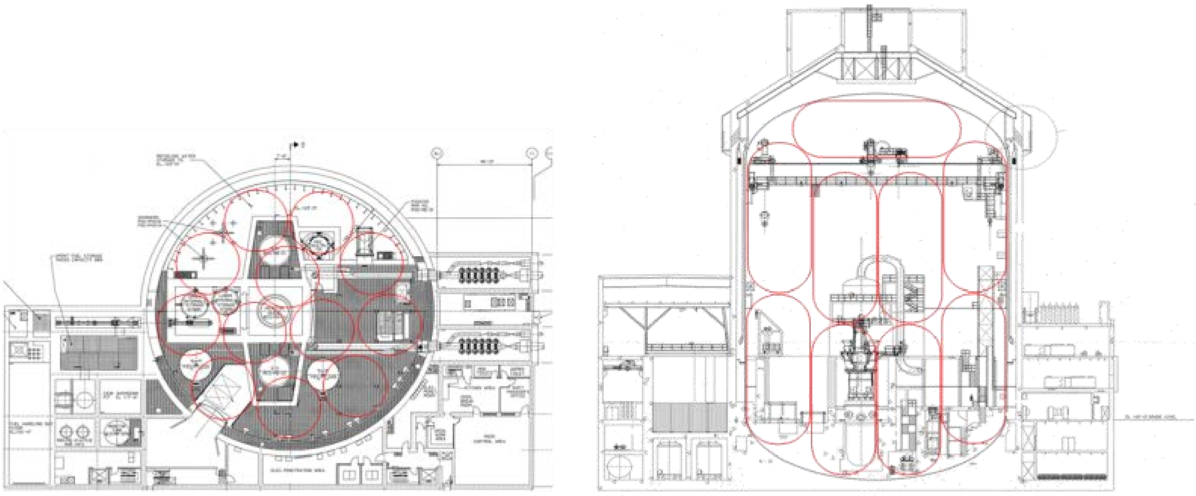
Specific Design Features

The elimination of CRDM penetrations through the RPV head prevents postulated rod ejection accidents as well as potential nozzle cracking that adversely impacted currently licensed plant designs. Eight seal-less canned motor pumps are mounted horizontally to the shell of the RPV and provide forced reactor coolant flow through the core. The steam generator is a straight tube configuration with the primary reactor coolant passing through the inside of the tubes and the secondary coolant on the outside. An integral pressurizer located above the steam generator within the RPV is used to control pressure in the primary system. The moisture separation functions are relocated to a separate steam drum located outside of containment, reducing the reactor and containment vessel heights. The integral RPV has a diameter and height of 3.7 m and 28 m, respectively. Both the integral reactor vessel and the passive core cooling system are located within a compact, high pressure, steel containment vessel (CV) having a diameter and height of 9.7 m and 28.5 m, respectively. The CV operates at a vacuum, and is designed to be fully submerged in water to facilitate heat removal during accident events while providing for an additional radionuclide filter. Soluble boron is used in the reactor coolant system (RCS) for normal reactivity depletion, and control rods for load follow and plant shutdown. Safety injection, passive boration and heat removal are provided by the passive core cooling system and the ultimate heat sink.

Safety Features

The Westinghouse SMR is an advanced passive plant, where the safety systems are designed to mitigate accidents through the use of natural driving forces such as gravity flow and natural circulation flow. The plant is not reliant on AC power or other support systems to perform its safety functions. The seven day minimum coping time following loss of offsite power is a fundamental advancement over the three day coping time of the best, currently licensed plants. The integral reactor design eliminates large loop piping and potential large break loss of coolant accidents (LOCA), and significantly reduces the flow area of postulated small break LOCAs. The below grade locations of the reactor vessel, containment vessel, and spent fuel pool provide protection against external threats and natural phenomena hazards. The small size and low power density of the reactor limits the potential consequences of an accident relative to a large plant. The plant is designed to be “standalone” with no shared systems, eliminating susceptibility to failures that cascade from one unit to another in a multi-unit station. The result is a plant capable of withstanding natural phenomena hazards and beyond-design-basis accident scenarios, including long-term station black-out (SBO).

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	Westinghouse Electric Company LLC
Country of Origin:	USA
Reactor Type:	Integral PWR
Electrical Capacity (MW(e)):	> 225
Thermal Capacity (MW(th)):	800
Expected Capacity Factor (%):	95
Design Life (years):	60
Plant Footprint (m^2):	65 000
Coolant/Moderator:	Light water
Primary Circulation:	Forced circulation
System Pressure (MPa):	15.5
Main Reactivity Control Mechanism:	CRDM, boron
RPV Height (m):	28
RPV Diameter (m):	3.7
Coolant Temperature, Core Outlet (°C):	324
Coolant Temperature, Core Inlet (°C):	294
Integral Design:	Yes
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	No
Low-Temp Process Heat:	Possible
Cogeneration Capability:	Possible
Design Configured for Process Heat Applications:	No
Passive Safety Features:	Yes
Active Safety Features:	Active defence-in-depth functions only
Fuel Type/Assembly Array:	UO_2 pellet/17x17 square
Fuel Active Length (m):	2.4
Number of Fuel Assemblies:	89
Fuel Enrichment (%):	< 5
Fuel Burnup (GWd/ton):	> 62
Fuel Cycle (months):	24
Number of Safety Trains:	Three diverse decay heat removal methods
Emergency Safety Systems:	Passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	17
Distinguishing Features:	Incorporates passive safety systems and proven components of the AP1000 plant
Modules per Plant:	1
Estimated Construction Schedule (months):	18 – 24
Seismic Design:	Based on CEUS sites
Predicted Core Damage Frequency (per reactor year):	$< 5 \times 10^{-8}$
Design Status:	Preliminary design completed

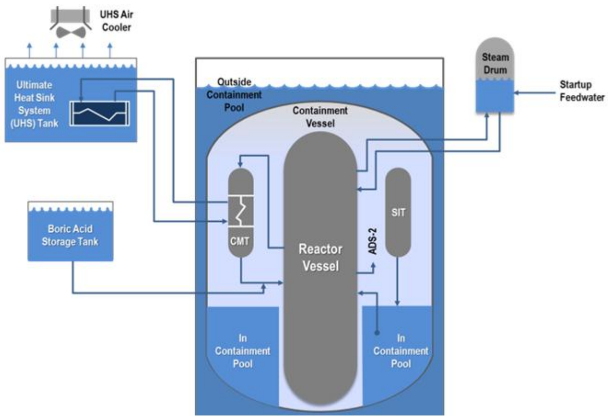


Comparison of containment volumes between Westinghouse AP1000 plant and SMR plant

Three diverse decay heat removal methods are provided in the Westinghouse SMR. The first method of decay heat removal uses gravity feed from the steam drum through the steam generator for approximately 80 minutes of natural circulation cooling. In this scenario, steam is released to the atmosphere through two redundant power-operated relief valves (PORV).

The second decay heat removal method can be achieved by cooling the RCS with a passive decay heat removal heat exchanger, one of which is located in each of four core makeup tanks (CMT). Heat from the CMTs is then rejected to four heat exchangers located in two ultimate heat sink system (UHS) tanks. The UHS tanks are sized to provide a minimum of seven days of decay heat removal, with additional options to replenish lost inventory and cool the plant indefinitely.

A third diverse method of decay heat removal capability is available by cooling the RCS with diverse bleed and feed methods including a two-stage automatic depressurization system (ADS) that vents the RCS to the containment through direct vessel injection (DVI) pathways, water injection from the four CMTs and in-containment pool (ICP) tank paths, and gravity-fed boric acid tank water makeup to the DVI paths. The steam vented from the RCS to the containment is cooled and condensed by the containment shell. The containment shell is cooled by the water in the outside containment pool (OCP) which completely surrounds the containment. When the OCP water eventually boils, makeup water is provided by gravity from each of the two redundant UHS tanks that maintain the OCP full of water. The water condensed on the containment shell flows back into the RCS through two sump injection flow paths.



Three diverse decay heat removal methods of the Westinghouse SMR

Electrical, and Instrumentation and Control Systems

An Ovation® -based digital I&C system controls the normal operations of the plant. The protection and safety monitoring system (PMS) provides detection of off-normal conditions and actuation of appropriate safety-related functions necessary to achieve and maintain the plant in a safe shutdown condition. The plant control system (PLS) controls non-safety-related components in the plant that are operated from the main control room or remote shutdown workstation. The diverse actuation system (DAS) is a non-safety-related, diverse system that provides an alternate means of initiating reactor trip and actuating selected engineered safety features.

The Westinghouse SMR onsite power system consists of a main AC power system and a DC power system. The main AC power system is a non-Class 1E system and does not perform any safety-related functions. The plant DC power system is comprised of independent Class 1E and non-Class 1E DC power systems. Safety-related DC power is provided to support reactor trip and engineered safeguards actuation. Batteries are sized to provide the necessary DC power and uninterruptible AC power for items such as PMS actuation; control room functions, including habitability; DC-powered valves in the passive safety-related systems; and containment isolation. Two diverse, non-safety AC power backup systems are provided: 1) diesel-driven generators to provide power for defence-in-depth electrical loads, and 2) a decay heat-driven generator. The decay heat-driven generator provides AC power to the plant using the heat generated by the core following reactor trip.

Description of the Turbine-Generator Systems

The steam turbine of the Westinghouse SMR operates at 3600 RPM. The electrical generator is designed for air cooling, eliminating the potential for explosions that can occur with hydrogen-cooled options. The Westinghouse SMR condenser design includes the ability to use air-cooling. The turbine is designed to accommodate a wide variety of backpressures with different blade configurations optimized for narrow-range, high performance power. The water intake requirements will be comparable to existing plants on a per-power basis, but significantly less on a plant basis because of the lower power rating. This low water usage enables the reactor to be sited in places previously not available for nuclear construction.

Fuel Characteristics and Fuel Supply Issues

The Westinghouse SMR reactor core is based on the licensed Westinghouse robust fuel assembly (RFA) design, and uses 89 standard 17 x 17 fuel assemblies with an 2.4 m active fuel height and Optimized ZIRLO™ cladding for corrosion resistance. A metallic radial reflector is used to achieve better neutron economy in the core while reducing enrichment requirements to less than the existing statutory limit of 5.0 wt% ^{235}U. Approximately 40% of the core is replaced every two years, resulting in an efficient and economical operating cycle of 700 effective full power days (EFPD) that coincides with existing regulatory surveillance intervals.

Below grade location of the reactor and containment vessels of the Westinghouse SMR

Westinghouse SMR conceptual site layout

Licensing and Certification Status

Westinghouse is considering a number of business models for the successful deployment of the Westinghouse SMR product globally. Most of these models assume varying levels of cost-sharing with industry consortium and/or government partners for First-of-a-Kind (FOAK) development and licensing costs to advance the Westinghouse SMR towards early site permits and design certification.

All the figures provided courtesy of Westinghouse Electric Company, LLC, with permission.

SMR-160 (Holtec International, USA)

Introduction

The SMR-160 conceptual design has been developed by Holtec International as an advanced PWR-type small modular reactor producing power of 525 MW(th) or 160 MW(e) adopting passive safety features. Simplification in the design is achieved by using fewer valves, pumps, heat exchangers, instrumentation, and control loops than conventional plants, simplifying operator actions during all plant modes, including diagnosing and managing off-normal and accident conditions. The SMR-160 uses fuel very similar to existing commercial LWR product lines, includes no reactor coolant pumps and utilizes a large vertical steam generator. A modular construction plan for SMR-160 involves pre-assembling the largest shippable components prior to arrival at a site. A 24-month construction period is envisaged for each unit.

Reactor System Configuration of SMR-160

Development Milestones

2015	Complete conceptual design

Target Applications

The primary application of SMR-160 is electricity production with optional cogeneration equipment (i.e., hydrogen generation, district heating, and seawater desalination). Target applications include distributed electricity production, repowering coal facilities, uprate existing nuclear facilities and providing electricity and low temperature process heat for commercial and military installations. Design optimization includes air cooled condensation for no wet cooling.

Specific Design Features

The SMR-160 is a pressurized water reactor with reactor coolant system (RCS) using natural circulation for all power and accident modes and states. The RCS is comprised of the reactor pressure vessel (RPV) and a steam generator (SG) in an offset configuration with an integrated pressurizer flanged to the top of the steam generator. The RPV and the SG are connected by a single connection which houses both the hot leg and the cold leg. The hot leg is the inner pipe and the cold leg is the annular region of this single connection. Unique among integrated PWRs, the offset configuration allows easy access to the core without moving the RPV or SG during refuelling. The SG has a superheating feature which eliminates the need for a moisture separator reheater (MSR) and trains of feedwater heaters while not compromising the thermodynamic efficiency of the plant.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	Holtec International
Country of Origin:	USA
Reactor Type:	PWR
Electrical Capacity (MW(e)):	160
Thermal Capacity (MW(th)):	525
Expected Capacity Factor (%):	> 98
Design Life (years):	80
Plant Footprint (m^2):	20 500
Coolant/Moderator:	Light water
Primary Circulation:	Natural circulation
System Pressure (MPa):	15.5
Main Reactivity Control Mechanism:	CRDM only
RPV Height (m):	15
RPV Diameter (m):	2.7
Coolant Temperature, Core Outlet (°C):	316
Coolant Temperature, Core Inlet (°C):	196
Integral Design:	Yes
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	No
Low-Temp Process Heat:	Possible
Cogeneration Capability:	Possible
Design Configured for Process Heat Applications:	No
Passive Safety Features:	Yes
Active Safety Features:	No
Fuel Type/Assembly Array:	UO_2 pellet/ square array
Fuel Active Length (m):	3.7
Number of Fuel Assemblies:	N/A
Fuel Enrichment (%):	4.95
Fuel Burnup (GWd/ton):	32 GWd/t (initial design)
Fuel Cycle (yrs.):	3 – 4
Number of Safety Trains:	2
Emergency Safety Systems:	Numerous, all passive
Residual Heat Removal Systems:	2 active, 1 passive (safety)
Refuelling Outage (days):	5
Distinguishing Features:	• Unique approach to Defence-in-Depth with active non-safety and passive safety cooling systems • Unique, revolutionary integrated fuel cartridge provides: ○ Simplified fuelling/refuelling ○ Integrated, indefinite, passive spent fuel cooling ○ Integrated spent fuel dry storage
Modules per Plant:	1
Estimated Construction Schedule (months):	24
Seismic Design:	Robust
Predicted Core Damage Frequency:	Tiny
Design Status:	Conceptual design

The secondary side has only one feedwater heater simplifying plant operations and maintenance. The RPV is located in a free standing steel containment vessel called the containment structure (CS), supported within a reinforced concrete reactor building called the containment enclosure structure (CES) which also provides missile protection. The annular region between the CS and the CES also called the coolant reservoir (CR) is filled with water and serves as the Ultimate Heat Sink (UHS) for SMR-160.

The control rod drive mechanism (CRDM) based on existing technology is located outside the reactor coolant system on the RPV top head. The pressurizer uses heaters and cold-water injection nozzles to perform the same functions of a typical pressurizer. Integrating the pressurizer with the steam generator eliminates the typical primary cold- and hot-legs along with their supporting structures normally connecting the primary external loop of a PWR to an external pressurizer and reactor coolant pumps. All of the electrical connections required to run the CRDMs and the pressurizer's heaters are external resulting in a highly simplified design. All of the electrical connections required to run the CRDMs and the pressurizer's heaters are external the resulting design is highly simplified. The underground containment vessel part of the SMR-160 design houses the RPV and sections of the unitized integral steam generator unit, and the above ground part houses the top sections of the integral steam generator and pressurizer unit. The underground containment vessel part of the SMR-160 design is housing the RPV and sections of the unitized integral steam generator unit, and the above ground part is housing the top sections of the integral steam generator-and pressurizer unit.

Containment structure and containment enclosure structure

Safety Features

The RPV is located entirely below grade providing physical protection against aircraft hazards or missiles. The water filled Coolant Reservoir (UHS) provides the means by which the SMR-160 passively cools the reactor and spent fuel following a design basis event. SMR-160 relies on gravity-driven passive safety systems to ensure the plant is maintained in a safe configuration for all postulated design basis and beyond design basis events.

Passive decay heat removal from core and spent fuel pool is executed through heat exchange with dedicated passive heat exchangers which subsequently deliver the heat to the annular water body through the Containment Structure metal. Long-term decay heat removal from core and spent fuel pool is achieved (once the water in the UHS has evaporated) by air cooling enhanced by fins on the external surface of the CS thus assuring that heat removal will continue indefinitely.

Passive decay heat removal from core and spent fuel pool is executed through heat exchange with a dedicated passive heat exchangers which subsequently deliver the heat to the annular water body through the Containment Structure metal. Long-term decay heat removal from core and spent fuel pool is achieved(once the water in the UHS has evaporated) by air cooling enhanced by fins on the external surface of the CS thus assuring that heat removal will continue indefinitely.

In the event of abnormal transients or postulated accidents, SMR-160 will employ non-safety active cooling systems as the first line of defence, failure of which will actuate the passive safety systems and the plant will shutdown and remain safely shutdown for an unlimited period without the need for power, make-up water or operator actions.

The reactor core and spent fuel pool would remain safely cooled under a loss-of-coolant accident (LOCA); break in a main steam line; or prolonged loss of power. All reactor coolant system piping of more than 8 inches in diameter are eliminated, thus increasing the plant's ability to withstand a LOCA. SMR-160 safety is not affected by loss of offsite power (LOOP) or station black-out (SBO) events.

The core is configured as a cartridge with optimized flow paths to reduce flow induced stresses on fuel rods, and is located at the bottom of an extended pressure vessel containing a large coolant inventory above the core. Therefore, even under an improbable large break LOCA the core remains covered by a large inventory of water.

The SMR-160 passive design eliminates potential accidents and events that could cause core damage or precipitate releases into the environment. The plant is designed to withstand the effects of all natural phenomena including earthquakes and severe weather events.

SMR-160's safety design features also include passive autocatalytic hydrogen recombiners to mitigate potential hydrogen build-up caused by a metal to water reaction in the core.

Fuel Characteristics and Fuel Supply Issues

Full-length fuel assemblies with variable enrichments up to 4.95% are located in a Unitary Fuel Cartridge, which is loaded and unloaded as a single unit. Control is provided by reactivity, axial power-shaping, and shutdown control rods respectively, without soluble boron.

Licensing and Certification Status

The pre-application activities for the technology have started with the US-NRC. The Project baseline plan reflects realistic work scope, task durations and schedules to ensure Design Certification in time to support commercial operation of the first plant by 2025.

All the figures provided courtesy of Holtec International, with permission.

HIGH TEMPERATURE GAS COOLED REACTORS

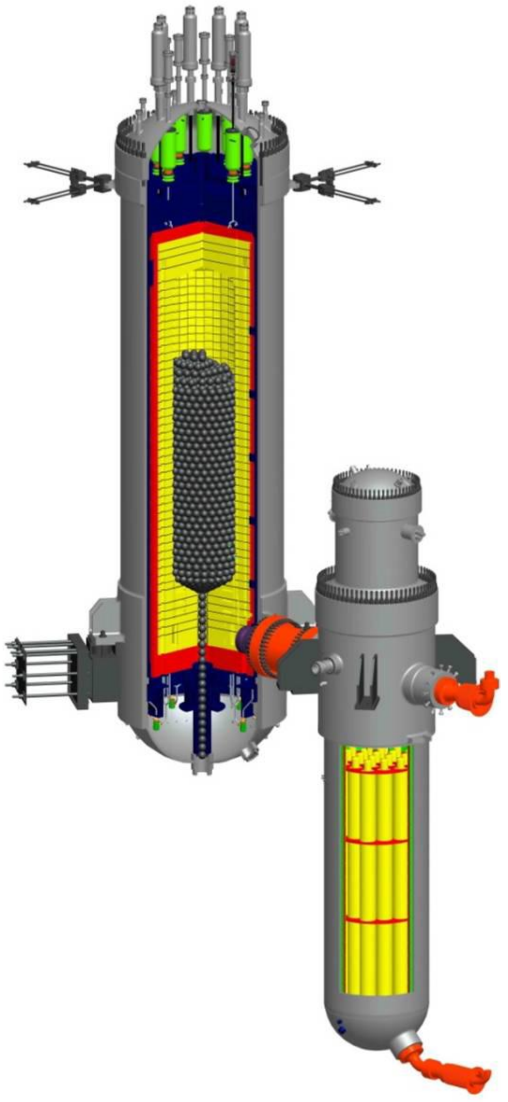
HTR-PM (Tsinghua University, China)

Introduction

In March 1992, the China Central 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) operation. Tsinghua University's Institute of Nuclear and New 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;
- Withdrawal of control rod without any countermeasures;
- Helium blower trip without closing outlet cut-off valve.

The second step of HTGR application in China began in 2001 when the HTR-PM project was launched



Reactor Configuration of HTR-PM
(Courtesy of Tsinghua University, with permission)

Development Milestones

1995	Start construction of HTR-10
2000	HTR-10 first criticality
2001	Launch of commercial HTR-PM project
2003	HTR-10 full power operation
2004	HTR-PM standard design was started jointly by INET and Chinergy Co.
2006	Project approved as national key technology project
2006	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
2008	HTR-PM Basic design completed
2009	Assessment of HTR-PM PSAR completed
2012	Helium Test loop construction completed
2012	HTR-PM First Pour of Concrete
2013	Fuel plant construction completed with installation of equipment on-going
2017	First operation expected

Target Applications

The HTR-PM is a commercial demonstration unit for electricity production. The twin reactor units driving a single turbine configuration was specifically selected to demonstrate its feasibility.

Following HTR-PM, commercial deployment of HTR-PM based on batch construction is foreseeing, and units with more modules and bigger power size are under investigation. Standardized reactor modules with 2, 6 or 9 reactors with a single turbine (200, 600 or 1000MW) are envisaged.

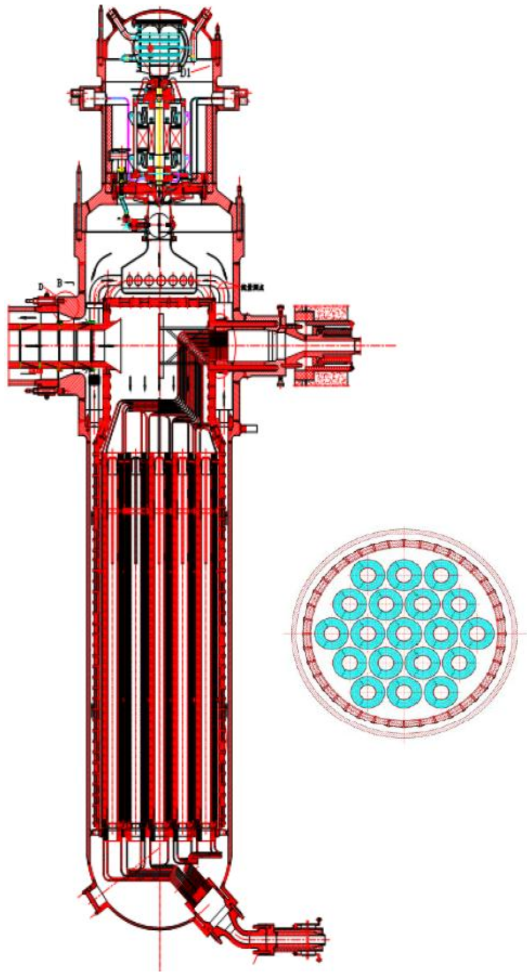
Development and research on process heat applications, hydrogen production and gas turbines are continuing for future application.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	Tsinghua University
Country of Origin:	China
Reactor Type:	Modular Pebble bed High Temperature Gas Cooled Reactor
Electrical Capacity (MW(e)):	210
Thermal Capacity (MW(th)):	2x250
Expected Capacity Factor (%):	85
Design Life (years):	40
Plant Footprint (m^2):	~60,000 (site)
Coolant/Moderator:	Helium / Graphite
Primary Circulation:	Forced circulation
System Pressure (MPa):	7
Main Reactivity Control Mechanism:	Negative temperature coefficient; control rod insertion
RPV Height (m):	25
RPV Diameter (m):	5.7 (inner)
Coolant Temperature, Core Outlet (°C):	750
Coolant Temperature, Core Inlet (°C):	250
Integral Design:	No
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	Yes, possible with different configuration
Low-Temp Process Heat:	Yes, possible with different configuration
Cogeneration Capability:	Electricity only; possible with different configuration
Design Configured for Process Heat Applications:	No
Passive Safety Features:	Yes, large negative temperature coefficients, large heat capacity
Active Safety Features:	Yes, control rod insertion with SCRAM; Turbine trip
Fuel Type/Assembly Array:	Pebble bed with coated particle fuel
Fuel Pebble Diameter (cm):	6
Number of Fuel Spheres:	420,000
Fuel Enrichment (%):	8.5
Fuel Burnup (GWd/ton):	90 (average discharge)
Fuel Cycle (months):	N/A; Online / on-power refuelling Fuel remain in the reactor for ~35 months
Number of Safety Trains:	N/A
Emergency Safety Systems:	Control rod insertion; Circulator trip; isolation of secondary circuit; drain of steam generator in the case of steam generator tube break
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	N/A since online
Distinguishing Features:	Passive post-shutdown decay heat removal
Modules per Plant:	Two reactors with their own steam-generator feeding one turbine-generator set
Estimated Construction Schedule (months):	59 months for 1st unit under construction
Seismic Design (g):	0.2
Predicted Large Release Frequency:	Core damage frequency not applicable to HTGRs No off-site shelter or evacuation plan needed
Design Status:	Under construction

Specific Design Features

The primary circuit consists of the reactor pressure vessel, the steam generator (SG) pressure vessel and the hot gas duct vessel connecting the two in a side-by-side arrangement. The core is a ceramic cylindrical shell housing the pebble bed, which acts as a reflector, heat insulator and neutron shield. The control rod system and the small absorber sphere system are two independent control systems of reactivity. These two independent systems fulfil the requirements of diversity and redundancy, and are both located in the reflector region. On average fuel spheres are circulated 15 times through the reactor to ensure a low power peaking factor and to limit the peak fuel temperatures during loss of coolant events. The design allows load follow (100%-50%-100%) but the two reactor units can further also be operated independently.

The main helium blower, designed as a vertical structure, is installed on the top of the steam generator inside the steam generator pressure vessel. The super-heated SG is a vertical, counter-flow, once-through generator with a helium–water interface. There are multiple units consisting of helical heat transfer tubes.

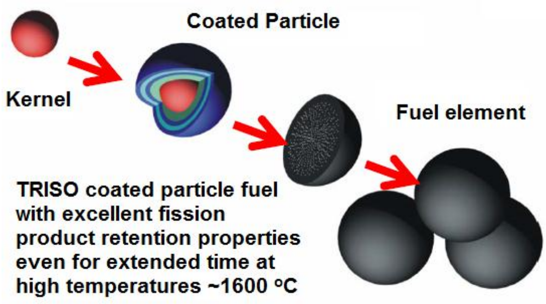


Once-through helical steam generator

Safety Features

The HTR-PM incorporates the inherent safety principles of the modular HTGR. The lower power density, good coated particle fuel performance and a balanced system design ensures that the fundamental safety functions are maintained. A large negative temperature coefficient, large temperature margin, low excess reactivity (on-line refuelling) and control rods ensures safe operation and limits accident temperatures. The decay heat is passively removed from the core under any designed accident conditions by natural mechanisms, such as heat conduction or heat radiation, and keeps the maximum fuel temperature below 1620°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.

Consequently there is no need for emergency core cooling system(s) in the design, and the decay heat is removed by natural mechanisms. The reactor cavity cooling system operates using cooling panels connected to an air cooler, but the fuel temperature will not exceed the limit if this system is not operating. The margins during normal operation and accident conditions are large (several hundred degrees in temperature).

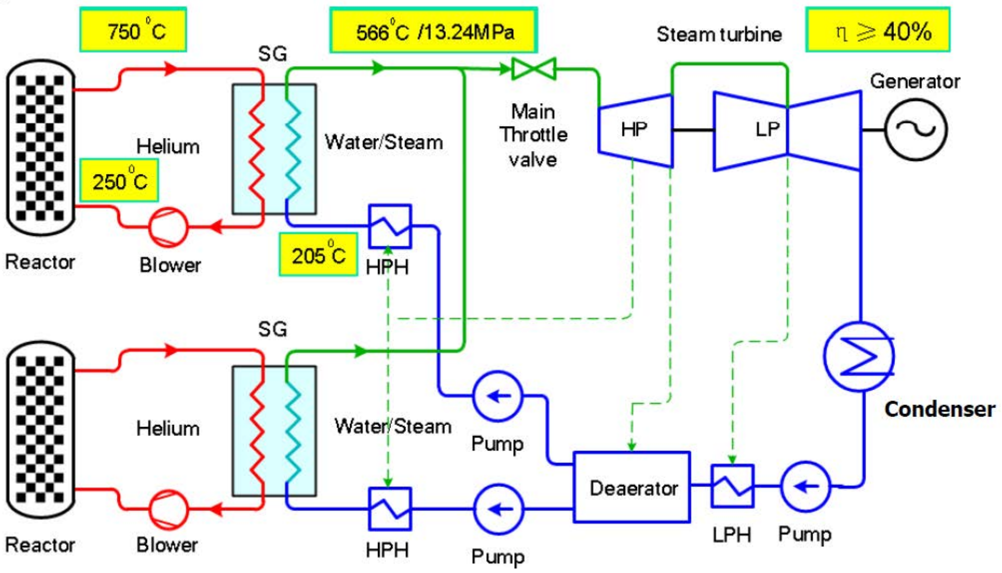


Fuel Characteristics

The HTR-PM utilizes the triple coated isotropic (TRISO) ceramic coated particle fuel element, which contains fuel kernels of 200–600 µm UO_2, UC2 and UCO, but can also contain thorium or plutonium. The various layers of the TRISO fuel element include a porous carbon buffer layer and two dense pyrolytic carbon layers separated by a silicon carbide layer.

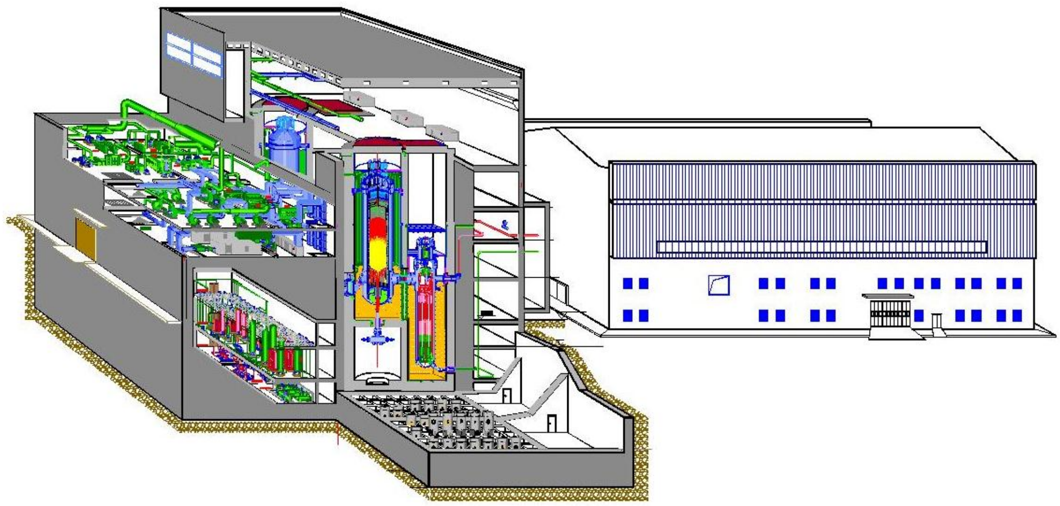
Description of the power conversion system

The Rankine power conversion with two reactor units and a single turbine-generator set is shown below. Each reactor unit has its own SG. During normal operation, driven by the primary helium circulator, the helium coolant passes the steam generator, and to heat up the 205°C feed water into 566°C superheated steam in the secondary loop. Then the superheated steam comes into the turbine to generate electricity. In the normal shutdown conditions, the reactor coolant system can also remove the core decay heat to the heat sink through the steam generator and the startup/shutdown circuit.



HTR-PM twin unit power conversion components and flow diagram (Source: Tsinghua University)

The nuclear island contain the two reactor units, and the building provide protection from external events. Other equipment for functions such as the fuel handling (loading, burnup measurement, unloading), the primary pressure release system, the secondary circuit isolation system, steam generatior emergency draining system, and the sub-pressure ventilation system are also loacted in the reactor building. The conventional island thermodynamic system consists of the condenser, main water feed water system, regenerative extraction steam system, heater drain deflation system, auxiliary steam system, plant recycled water and the open cycle cooling water systems, closed cycle cooling water systems, vacuum systems, etc.



HTR-PM nuclear island and conventional island layout (Source: Tsinghua University)

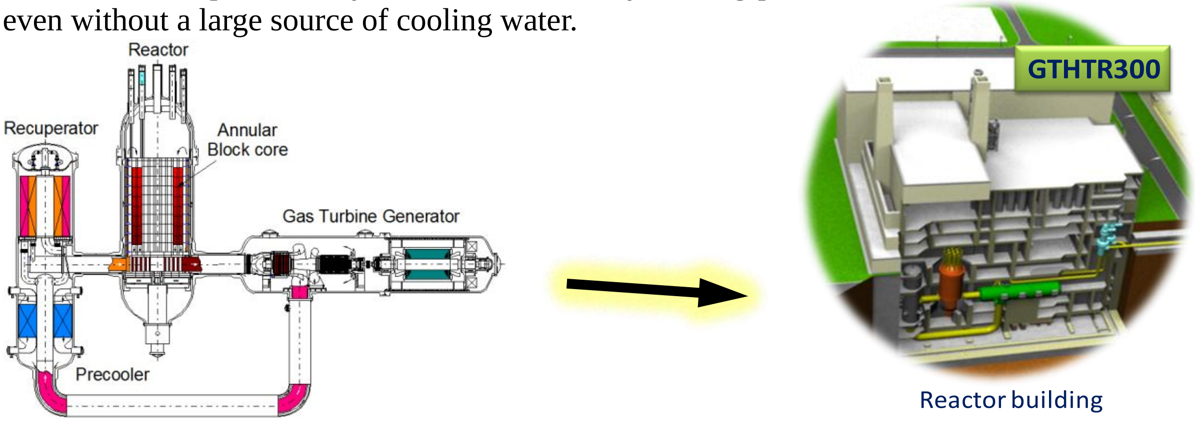
Licensing and Certification Status

The preliminary safety analysis report (PSAR) was accepted by the licensing authorities during 2008-2009. First concrete was poured in December 2012 and construction is progressing as planned. The FSAR (Final SAR) assessment is expected in 2016 with operation towards the end of 2017.

GT-HTR300 (Japan Atomic Energy Agency, Japan)

Introduction

GTHTR300 (Gas turbine High Temperature Reactor 300 MW(e) is a multipurpose, inherently-safe and site-flexible SMR (small modular reactor) that Japan Atomic Energy Agency is developing for commercialization in 2020s. As a Generation-IV technology, the GTHTR300 offers important advances comparing to current light water reactors. The reactor coolant temperature is significantly higher in the range of 850-950^{o}C. Such high temperature capability as proven in JAEA's HTTR test reactor operation enables a wider range of applications such as high temperature heat applications. The design employs direct-cycle helium gas turbine to simplify the plant by eliminating water and steam systems while delivering 45-50% generating efficiency comparing to about 33% efficiency by current reactors. The design incorporates all ceramic fuel, low power density but high thermal conductivity graphite core, and inert helium coolant to secure inherent reactor safety. The inherent safety permits siting proximity to customers, in particular to industrial heat users so as to minimize the cost and loss of high temperature heat supply. Dry cooling becomes economically feasible due to the use of gas turbine. The waste heat from the gas turbine cycle is rejected from 200^{o}C, creating large temperature difference from ambient air and making dry cooling tower size per unit of power generation comparable to the wet cooling towers used in nuclear plants today. The economical dry cooling permits inland and remote reactor siting even without a large source of cooling water.



Reactor System Configuration of GTHTR300

Development Milestones

2003	Basic design
2004	Design development start
2005	Cogeneration design (GTHTR300C)
2014	IS process continuous H2 production test facility construction
2015	Balance of plant connection to HTTR reactor start (planning)

Target Applications

Typical applications include electric power generation, thermochemical hydrogen production, desalination cogeneration using waste heat only, and steelmaking. The reactor thermal power may be rated up to 600 MWt maximum. The maximum product output per reactor is 120 t/d hydrogen enough to fuel about one million cars, 280-300 MW(e) electricity generation with additional seawater desalination cogeneration of 55,000 m^3/d potable water for about a quarter million of population, and annual production of 0.65 million tons of steel. All these are produced without CO_2 emission.

Plant arrangement for cogeneration applications (Courtesy of JAEA, with permission)

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	JAEA
Country of Origin:	Japan
Reactor Type:	Prismatic HTGR
Electrical Capacity (MW(e)):	100~300
Thermal Capacity (MW(th)):	< 600
Design Capacity Factor (%):	> 90
Design Life (years):	60
Coolant/ Moderator:	Helium/ Graphite
Primary Circulation:	Forced circulation
System Pressure (MPa):	7
Reactivity Control Mechanism:	Control rod
RPV Height (m):	23
RPV Diameter (m):	8
Coolant Temperature, Core Outlet (°C):	850-950
Coolant Temperature, Core Inlet (°C):	587-633
Integral Design:	No
Power Conversion Process:	Direct Brayton cycle
High Temperature Process Heat:	Yes
Low Temperature Process Heat:	Yes
Cogeneration Capability:	Yes
Design Configured for Process Heat Applications:	Yes
Safety Features:	Inherent
Fuel Type/Assembly Array:	UO_2 TRISO ceramic coated particle
Fuel Block Length (m):	1
Number of Fuel Columns in Core:	90
Average Fuel Enrichment (%):	14
Average Fuel Burnup (GWd/ton):	120
Fuel Cycle (months):	48
Number of Safety Trains:	4
Emergency Safety Systems:	Inherent
Residual Heat Removal Systems:	Inherent
Refuelling Outage (days):	30
Distinguishing Features:	Multiple applications of power generation, hydrogen production, process heat supply, steelmaking, desalination, district heating
Modules per Plant:	Up to 4 reactors
Estimated Construction Schedule (months):	24 – 36
Seismic Design (g):	> 0.18 (automatic shutdown)
Predicted Core Damage Frequency (per reactor year):	$< 10^{-8}$
Design Status:	Basic design

Specific Design Features

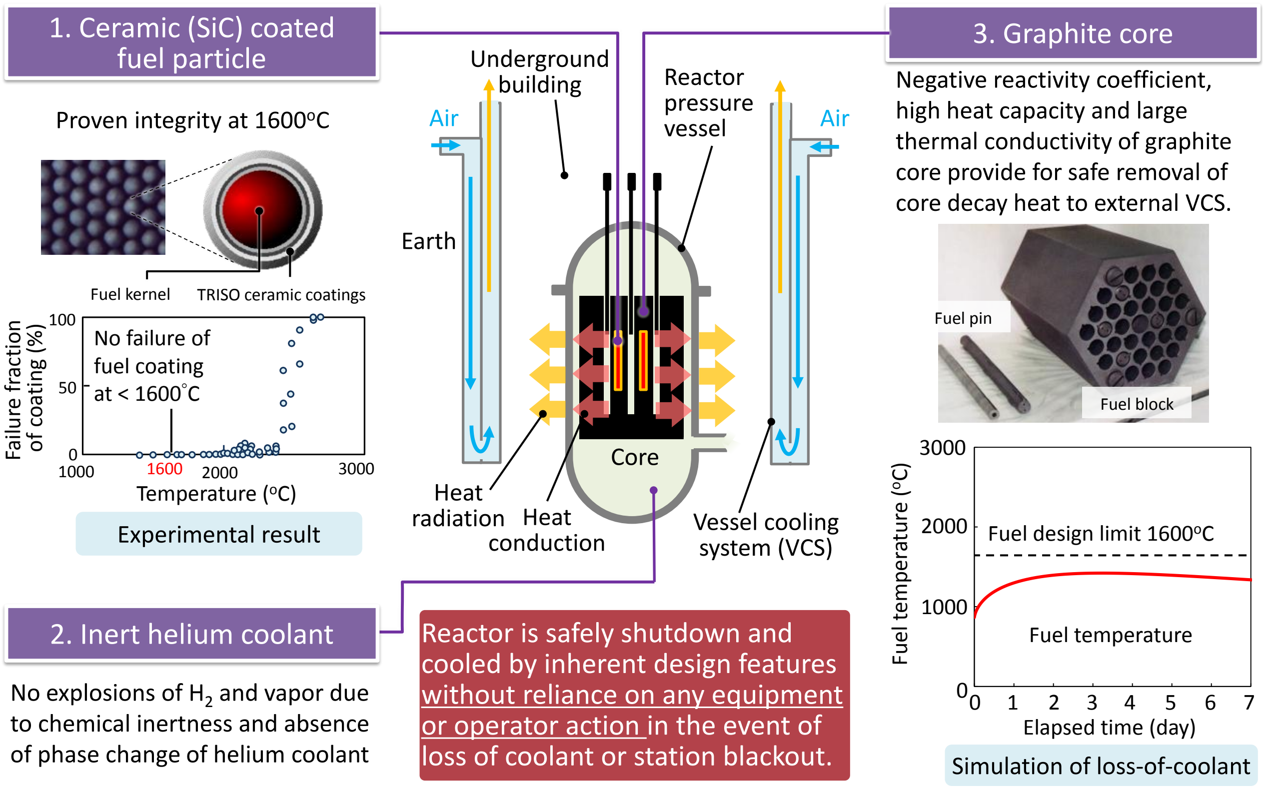
The reactor system combines a high temperature gas-cooled reactor with direct-cycle gas turbine to generate power while circulating the reactor coolant. The system consists of three functionally-oriented pressure vessel units, each housing reactor core, gas turbine, and heat exchangers respectively. The multi-vessel system facilitates modular construction and independent maintenance access to the functional vessel units. The reactor system is placed below grade in the reactor building. The pre-application basic design of the system was completed in 2003 by JAEA and domestic industrial partners Mitsubishi Heavy Industries, Fuji Electric, Nuclear Fuel Industries and others. The reactor system design added cogeneration capabilities in a GTHTR300C design that can accept various roles of cogeneration while sharing equipment designs with GTHTR300.

Safety Features

The reactor delivers fully inherent safety due to three enabling design features:

- The ceramic coated particle fuel maintains its containment integrity under the design temperature limit of 1600°C.
- The reactor helium coolant is chemically inert and thus absent of explosive gas generation or phase change.
- The graphite-moderated reactor core is designed having characterises of negative reactivity coefficient, low-power density and high thermal conductivity.

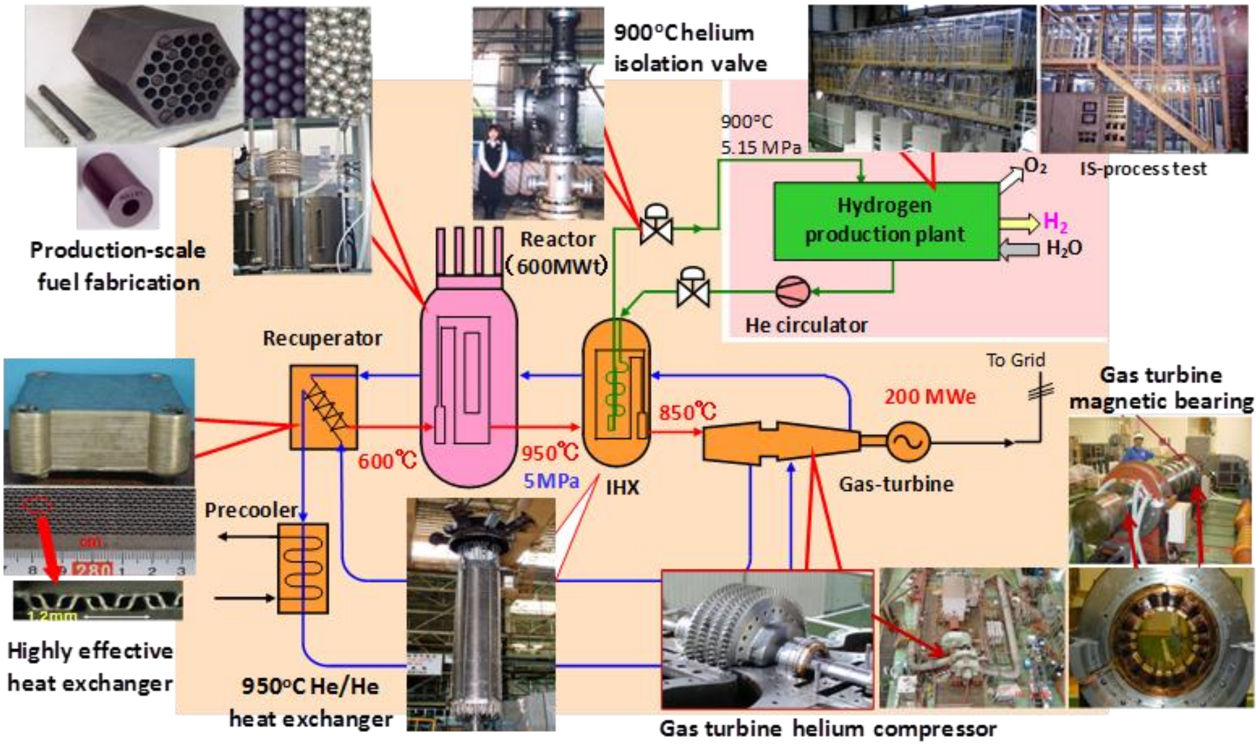
As a result of these features, the reactor core can be removed of decay heat by natural draft air cooling from outside of the reactor vessel for a period of days or months without reliance on any equipment or operator action even in such severe accident cases as loss of coolant or station blackout, where the fuel temperature will remain below the fuel design limit.



Inherent reactor safety design (Courtesy of JAEA, with permission)

Balance of plant system

While the reactor technologies required for the GTHTR300 are developed mainly with construction and operation of JAEA's 30 MWt and 950°C test reactor, we are separately developing and testing all key balance of plant technologies needed for the commercial SMR, including test validation of the helium gas turbine equipment at one-third to full scale, production-scale fuel fabrication lines, thermochemical hydrogen production process, and the superalloy heat exchanger capable of transferring 950°C reactor heat to the hydrogen production process.



Development for balance of plant technologies (Courtesy of JAEA, with permission)

Fuel Characteristics and Fuel Supply Issues

The fuel design employs spherical particles of less than 1 mm in diameter. Each particle consists of a UO_2 kernel coated by four layers of low and high density pyro-carbon and silicon carbide. The all ceramic particle fuel is heat resistant up to 1600°C. Approximately ten-thousand particles are packaged into a compact of the size of a thumb. The compacts are then assembled into graphite-clad fuel rods. The fuel rods are inserted into the bore holes of a hexagonal graphite fuel block of about 1 m long and 41 cm across, where the annulus formed between the fuel rod and the bore hole provides coolant flow channels. The fuel blocks are loaded into the reactor core. The more fuel blocks are placed in the core, the higher the power output of the reactor.

Licensing and Certification Status

The design is developed at pre-licensing basic design stage. The design and development are planned to be concluded to prepare for the lead plant construction around 2025.

(All figures provided courtesy of Japan Atomic Energy Agency, with permission)

GT-MHR (OKBM Afrikantov, Russian Federation)

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 the reactor unit 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.

The use of modular helium reactor units makes the system flexible and allows the possibility to use various power unit schemes: with gas turbine cycle, steam-turbine cycle and with the circuit supplying high-temperature heat to industrial applications. The modular high temperature gas cooled reactor unit possess salient safety features with passive decay heat removal providing a high level of safety even in case of total loss of primary coolant.

The modular helium reactor design proved the possibility of unit modularity with a wide power range of a module from 200 to 600 MW(th) and NPP power variation as a function of module number. This provides good maneuvering characteristics of the reactor plant (RP) for regional power sources.

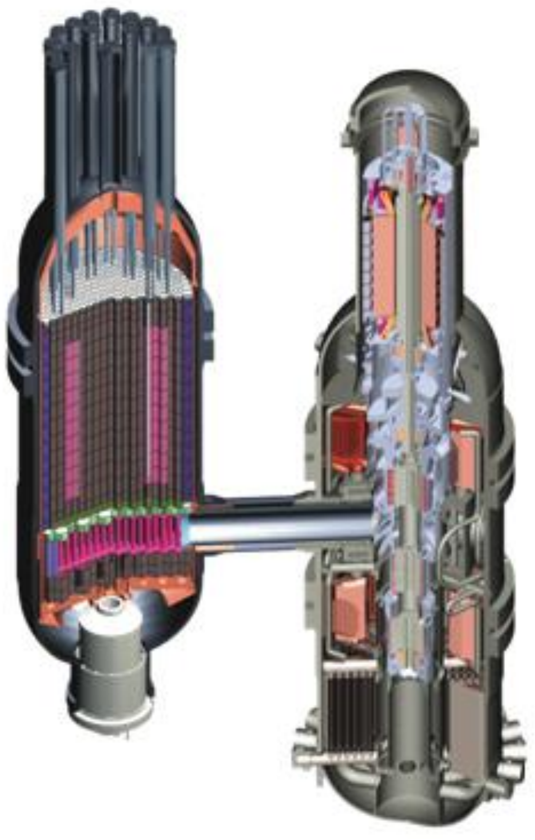
Development Milestones

1993	Minatom / General Atomics MOU on joint GT-MHR development for commercial units
1994	Russia proposes to build GT-MHR at Seversk to burn Russian WPu
1996	Framatome & Fuji Electric join the GT-MHR program
1997	Conceptual Design completed
1998	GT-MHR becomes an option within the US/RF Pu Disposition Strategy
1999	Conceptual Design review by international group of experts
2000	Work started on Preliminary Design
2002	Project review by Minatom of Russia and US DOE experts
2002	Reactor Plant Preliminary Design completed
2003	Begin Demonstration of key technologies

Target Applications

The GT-MHR can produce electricity at high efficiency (approximately 48%).

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.



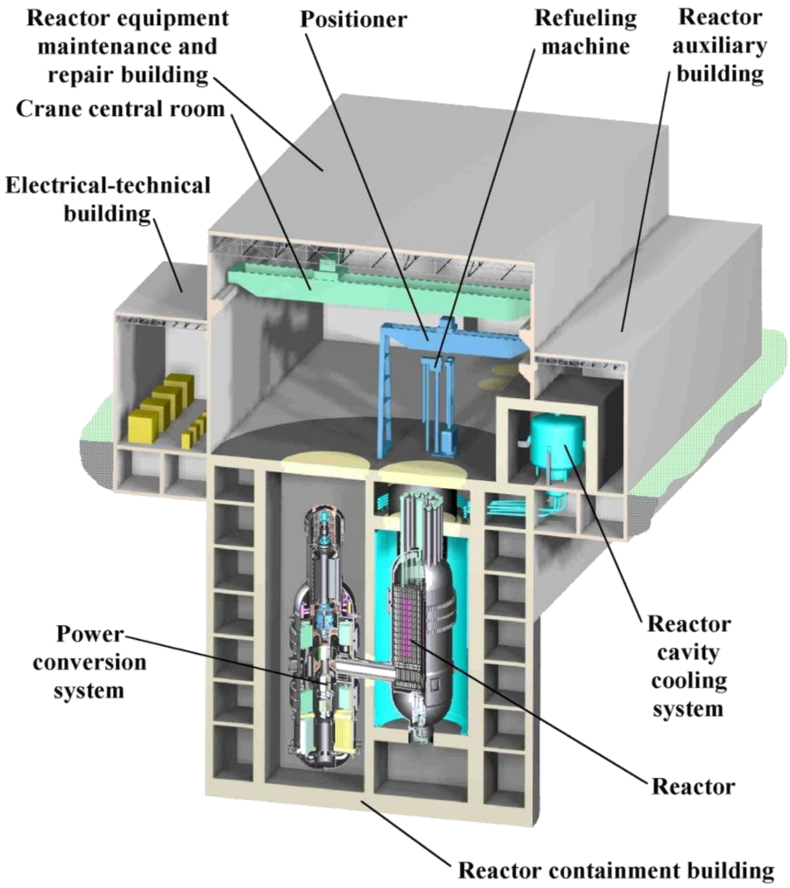
Reactor System Configuration of GT-MHR

(Courtesy of OKBM Afrikantov, with permission)

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	OKBM Afrikantov
Country of Origin:	Russian Federation
Reactor Type:	Modular helium reactor
Electrical Capacity (MW(e)):	285
Thermal Capacity (MW(th)):	600
Expected Capacity Factor (%):	80
Design Life (years):	60
Plant Footprint (m^2):	91100
Coolant/Moderator:	Helium
Primary Circulation:	Forced circulation
System Pressure (MPa):	7.24
Main Reactivity Control Mechanism:	Rod insertion
RPV Height (m):	31
RPV Diameter (m):	10.2
Coolant Temperature, Core Outlet (°C):	850
Coolant Temperature, Core Inlet (°C):	490
Integral Design:	Side-by-side Reactor and integral Gas Turbine
Power Conversion Process:	Direct Brayton or Indirect Rankine Cycle
High-Temp Process Heat:	Yes
Low-Temp Process Heat:	Yes
Cogeneration Capability:	Yes
Design Configured for Process Heat Applications:	Yes
Passive Safety Features:	Yes
Active Safety Features:	Yes
Fuel Type/Assembly Array:	Hexagonal Prism graphite block of 0.36m coated particle fuel
Fuel Active Length (m):	~0.8 m per block; 10 fuel blocks thus ~8 m active length
Number of Fuel Assemblies:	~1020 fuel blocks
Fuel Enrichment:	LEU or WPu
Fuel Burnup (GWd/ton):	100 – 720 (depending on the fuel cycle)
Fuel Cycle (months):	25
Number of Safety Trains:	2
Emergency Safety Systems:	Active and passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	25
Distinguishing Features:	Inherent safety characteristics; No core melt; High temperature process heat capabilities; Small number of safety systems
Modules per Plant:	1 (depends on application)
Estimated Construction Schedule (months):	48
Seismic Design:	8 points (MSK 64)
Predicted Large Release Frequency:	Core damage frequency not applicable to HTGRs BDBA frequency < 1 x10^{-5} /year Frequency of ultimate release at BDBA < 1 x10^{-7}/year
Design Status:	Conceptual design completed; Key technologies are being demonstrated

Specific Design Features

The GT-MHR direct Brayton cycle power conversion system contains a gas turbine, an electric generator and gas compressors. The GT-MHR gas turbine power conversion system has been made possible by utilizing large, active magnetic bearings, compact, highly effective gas to gas heat exchangers and high strength, high temperature steel alloy vessels. The use of the gas turbine cycle application in the primary circuit leads to a minimum number of reactor plant systems and components. 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 fuel form presents formidable challenges to diversion of materials for weapon production, as either fresh or as spent fuel.



GT-MHR layout (Courtesy of OKBM Afrikantov, with permission)

Safety Features

Safety is ensured by application of passive principles of system actuation. The decay and accumulated heat is removed from the core through reactor vessel to reactor cavity cooling system and then to atmosphere by natural physical processes of heat conductivity, radiation, convection without excess of fuel safe operation limits including LOCA, in case of all active circulation systems and power sources failure.

The design features, which determine the inherent safety of high temperature reactor, are the following:

- using of helium coolant, which has some specific properties. During plant operation, helium is not affected by phase transformations, does not dissociate, is not activated and has good heat transfer properties. Helium is chemically inert, does not react with fuel, moderator and structural materials. There are no helium reactivity effects;
- core and reflector structural material is high-density reactor graphite with substantial heat capacity and heat conductivity and sufficient mechanical strength that ensures core configuration preservation under any accident;
- nuclear fuel in the form of coated fuel particles with multilayer ceramic coatings, which retain integrity and effectively contain fission products under high fuel burnup and high

temperatures;

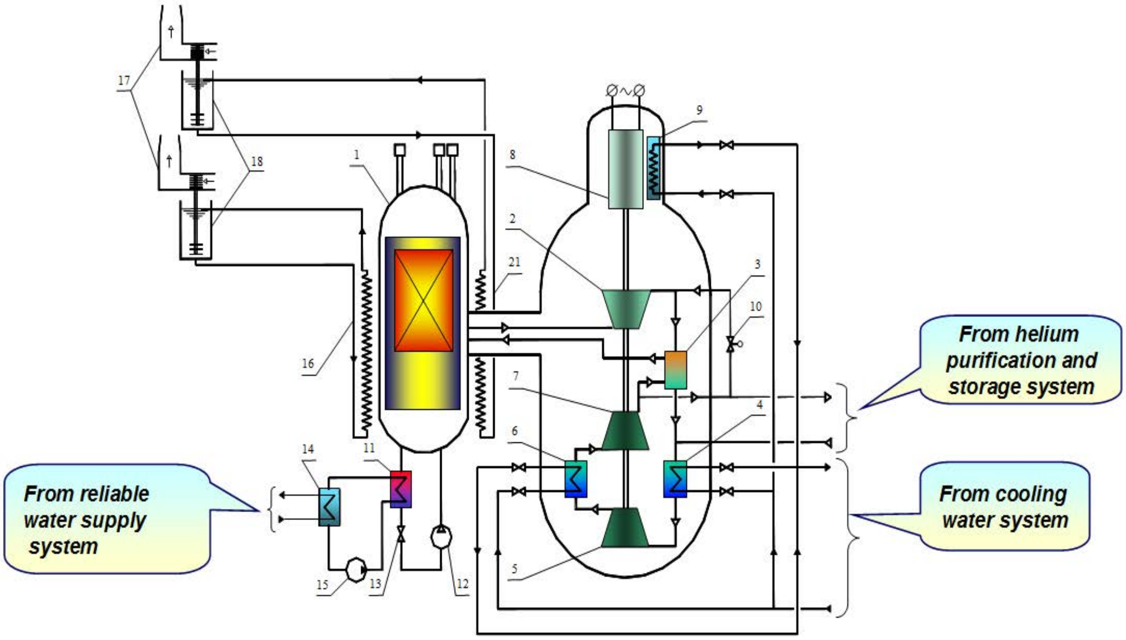
- as uranium fuel is used, temperature and power reactivity coefficients are negative that provides the reactor safety in any design and accident conditions.

Fuel Characteristics and Fuel Supply Issues

Coated particle fuel is used. The fuel kernel (U or Pu oxide) is coated by a first porous layer of pyrocarbon, followed by a dense layer of pyrocarbon, a silicon carbide layer and an outer dense layer of pyrocarbon. Thousands of coated particles and graphite matrix material are made into a fuel compact with thousands of compacts inserted into the fuel channels of the Hexagonal Prism graphite blocks or fuel assemblies. The coated particles will contain almost all fission products with temperatures up to 1600^{o}C. About 1 billion fuel particles of the same type were manufactured and tested in Russia. The standard fuel cycle for the commercial GT–MHR utilizes low enriched uranium (LEU) in a once through mode without reprocessing but alternative cycles including the disposition of plutonium were also studied in detail. The GT-MHR show good proliferation resistance characteristics. It produces less total plutonium and ^{239}Pu (materials of proliferation concern) per unit of energy produced. The GT-MHR fuel form presents formidable challenges to diversion of materials for weapons production, as either fresh or as spent fuel.

Description of the power conversion system

The Brayton power conversion with direct gas turbine is shown below.



GT-MHR flow diagram
(1)-reactor, (2)-recuperator, (4), (6)-precooler and intercooler,
(5), (7) - low-pressure and high-pressure compressors, (8) – generator, (9) – generator cooler, (10) – bypass valve, (11)…(15) – SCS components, (16) – surface cooler of reactor cavity cooling system (RCCS), (17) – air ducts, (18) – heat exchanger with heat pipes

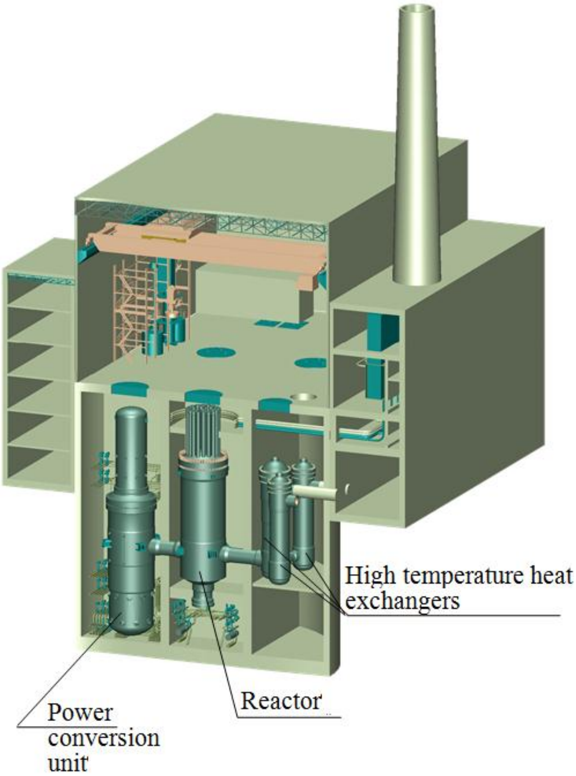
Licensing and Certification Status

Reactor Plant Preliminary Design completed with the demonstration of key technologies underway

MHR-T reactor/Hydrogen production complex (OKBM Afrikantov, Russian Federation)

Introduction

The MHR-T reactor/hydrogen production complex makes use of the basic GT-MHR reactor unit design as the basis for a multi-module plant for hydrogen production. The hydrogen production through the steam methane reforming process or high-temperature solid oxide electrochemical process is performed by coupled the plant with the modular helium reactor(s). The use of modular helium reactor units makes the system flexible and allows the possibility to use various power unit schemes: with gas turbine cycle (the GT-MHR), steam-turbine cycle and with the circuit supplying high-temperature heat to industrial applications (this design). The modular high temperature gas-cooled reactor unit possess salient safety features with passive decay heat removal providing a high level of safety even in case of total loss of primary coolant.



MHR-T RP layout – variant with steam methane reforming (OKBM Afrikantov)

Development Milestones

2001	Pre-conceptual proposal
2005	Conceptual design completed
2007	Elaboration of technical requirements

Target Applications

The most perspective technologies for Russia are hydrogen production through the steam methane reforming process or high-temperature solid oxide electro-chemical process coupled with a modular helium reactor called MHR-T. The chemical-technological sector with steam methane reforming is considered as an option for short-term perspective.

Specific Design Features

The MHR-T complex includes the chemical-technological sector (hydrogen production sector) and the infrastructure supporting its operation. The chemical-technological sector includes hydrogen production process lines, as well as systems and facilities supporting their operation. The following processes are considered as the basic processes for the chemical-technological sector: (i) steam methane reforming; and (ii) high-temperature solid oxide electrochemical process of hydrogen production from water. Heat shall be transferred directly from primary coolant to chemical-technological sector medium in a high-temperature heat exchanger. The key component of chemical-technological sector medium circulating through the high-temperature heat exchanger is water steam. The high-temperature electrolysis option allows the consideration of two- and three-circuit plant configurations. The technical concept is based on:

- modular helium-cooled reactors with typical high level of inherent safety;
- fuel cycle based on uranium dioxide in the form of multi-layer coated particles, high burnup and burial of fuel blocks unloaded from the reactor without any additional processing;
- electromagnetic bearings operating almost without friction and applied in various technical areas;
- highly efficient high-temperature compact heat exchangers, strong vessels made of heat resistant steel.

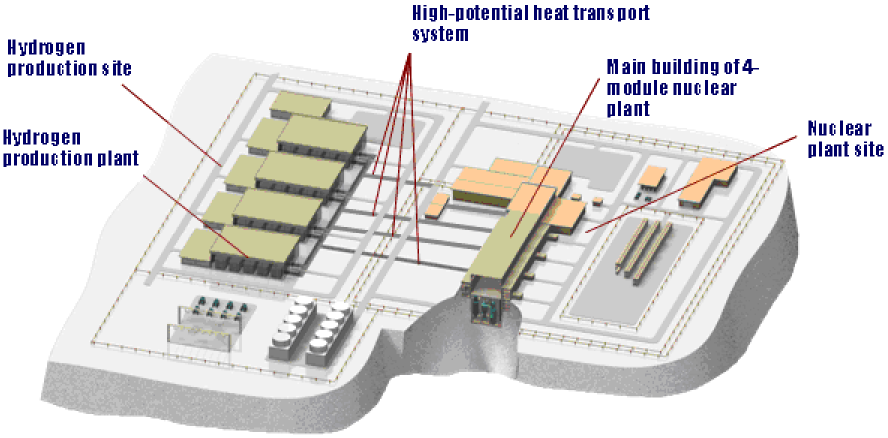
MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	OKBM Afrikantov
Country of Origin:	Russian Federation
Reactor Type:	Modular helium high temperature reactor
Electrical Capacity (MW(e)):	4x205.5
Thermal Capacity (MW(th)):	4x600
Expected Capacity Factor / Thermal Power Utilization (%):	80
Hydrogen or MHM Production Efficiency (gross), %	80
Hydrogen Production, (t/h):	4x12.5
Design Life (years):	60
Plant Footprint (m^2):	Not available
Coolant/Moderator:	Helium/graphite
Primary Circulation:	Forced circulation
System Pressure (MPa):	7.5
Main Reactivity Control Mechanism:	Rod insertion
RPV Height (m):	32.8
RPV Diameter (m):	6.9
Coolant Temperature, Core Outlet (°C):	950
Coolant Temperature, Core Inlet (°C):	578
Integral Design:	No
Power Conversion Process:	Direct closed Brayton cycle
High-Temp Process Heat:	Yes
Cogeneration Capability:	Yes
Design Configured for Process Heat Applications:	Direct supply of heat to the process of methane reforming
Passive Safety Features:	Yes
Active Safety Features:	Yes
Fuel Type/Assembly Array:	Hexagonal Prism graphite block of 0.36m coated particle fuel
Fuel Active Length (m):	~0.8 m per block; 10 fuel blocks thus ~8 m active length
Number of Fuel Assemblies:	~1020 fuel blocks
Fuel Enrichment:	20
Fuel Burnup (MWd/kg):	125
Fuel Cycle (months):	30
Number of Safety Trains:	2
Emergency Safety Systems:	Active and passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	25
Distinguishing Features:	Multi-module HTGR dedicated to hydrogen production / high temperature process heat application
Modules per Plant:	4
Estimated Construction Schedule (months):	48
Seismic Design:	8 points (MSK 64)
Predicted Core Damage Frequency:	Core damage frequency not applicable to HTGRs BDBA frequency < 1 $x10^{-5}$/year Frequency of ultimate release at BDBA < 1 $x10^{-7}$/year
Design Status:	Conceptual design

The main components of each NPP module are arranged in isolated premises of the underground containment of the NPP main building. The chemical-technological sector equipment is arranged outside the containment of the NPP main building. The MHR-T energy-technological complex is designed for a specific site on the basis of design solutions selected with account of:

- climatic conditions typical of central Russia;
- special external impacts – seismicity, aircraft crash, air shock wave.

The interfaces between the four-module NPP and the chemical-technological sector must be designed to except faults that could cause failure of more than one MHR-T module. The main reactor equipment is arranged in a vertical vessel located in a separate cavity parallel to the power conversion unit and high-temperature heat exchanger vessels.

The thermal energy generated in the reactor is converted to chemical energy in a thermal conversion unit (TCU) where, in the MHR-T option with methane reforming, the initial steam-gas mixture is converted to hydrogen-enriched converted gas (mixture of water steam, CO, H_2, CO_2, and CH_4) in the course of a thermochemical reaction.



MHR-T site with hydrogen production complex (Courtesy of OKBM Afrikantov)

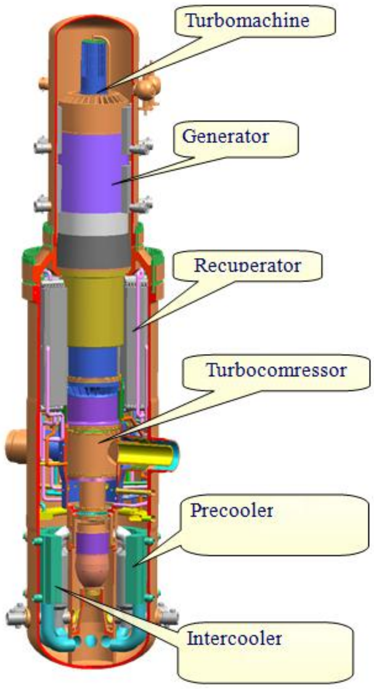
Safety Features

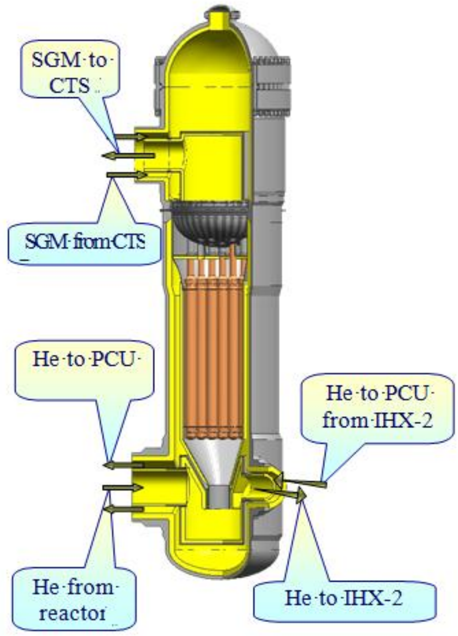
The safety features of the MHR-T reactor are the same as for the GT-MHR and it is thus repeated below. In addition special attention is given to external shock waves (due to an explosion at the chemical plant).

Safety is ensured by application of passive principles of system actuation. The decay and accumulated heat is removed from the core through reactor vessel to reactor cavity cooling system and then to atmosphere by natural physical processes of heat conductivity, radiation, convection without excess of fuel safe operation limits including LOCA, in case of all active circulation systems and power sources failure.

The design features, which determine the inherent safety of high temperature reactor, are the following:

- using of helium coolant, which has some specific properties. During plant operation, helium is not affected by phase transformations, does not dissociate, is not activated and has good heat transfer properties. Helium is chemically inert, does not react with fuel, moderator and structural materials. There are no helium reactivity effects;
- core and reflector structural material is high-density reactor graphite with substantial heat capacity and heat conductivity and sufficient mechanical strength that ensures core configuration preservation under any accident;
- nuclear fuel in the form of coated fuel particles with multilayer ceramic coatings, which retain integrity and effectively contain fission products under high fuel burnup and high temperatures;
- as uranium fuel is used, temperature and power reactivity coefficients are negative that provides the reactor safety in any design and accident conditions.





Power conversion unit *High temperature heat exchanger section*

Description of the Power Conversion System

A power conversion unit is integrated in a single vessel and includes a vertical turbomachine, highly efficient plate heat exchanger, and coolers.

The turbomachine includes a generator and turbocomressor mounted on a single shaft on electromagnetic suspension. Gas turbine cycle of power conversion with the helium turbomachine, heat exchanger and intercooler provides thermal efficiency at 48%.

A high-temperature heat exchanger (IHX) for the MHR-T option with methane steam reforming is an integral part of the thermal conversion unit and is partitioned as a three-stage heat exchanger. Arrangement of the heat exchanger sections along the primary coolant flow is parallel, and downstream of the coolant in the chemical-technological sector (CTS) is sequential. Each section is designed as a separate heat exchanger consisting of several modules. The material of the heat exchange surface of the module is a heat-resistant alloy.

Working media in circulation circuits are helium of the first circuit and steam-gas mixture (SGM) in the CTS. The peculiarity of heat exchangers for the production of hydrogen by methane reforming process is the transfer of heat from high temperature helium of the primary circuit to the chemically aggressive medium of hydrogen production circuit.

Fuel Characteristics and Fuel Supply Issues

Coated particle fuel is used. The fuel kernel from uranium oxide is coated by a first porous layer of pyrocarbon, followed by a dense layer of pyrocarbon, a silicon carbide layer and an outer dense layer of pyrocarbon. Thousands of coated particles and graphite matrix material are made into a fuel compact with thousands of compacts inserted into the fuel channels of the Hexagonal Prism graphite blocks or fuel assemblies. The coated particles will contain almost all fission products with temperatures up to 1600^{o}C. About 1 billion fuel particles of the same type were manufactured and tested in Russia. The standard fuel cycle is to utilize low enriched uranium (LEU) in a once through mode. The MHR-T show good proliferation resistance characteristics. It produces less total plutonium and ^{239}Pu (materials of proliferation concern) per unit of energy produced. The fuel form presents formidable challenges to diversion of materials for weapons production, as either fresh or as spent fuel.

Licensing and Certification Status

Conceptual Design completed.

MHR-100 (OKBM Afrikantov, Russian Federation)

Introduction

The design basis for these design elaborations is the world-wide experience in the development of experimental HTGR plants. Russia has experience (more than 40 years) in the development of HTGR plants of various power (from 100 to 1000 MW) and for various purposes, has established the experimental facilities for the R&D work, the fuel element and material fabrication technology, including the fabrication and mastering of pilot equipment, and various activities in hydrogen generation technology. Today, stations with electric capacity ~300 MWt, which are placed on all territory of Russia and adapted to regional systems, provide need of Russia in the electric power and are actually the regional power industry. The regional power industry consists mainly of cogeneration plants producing about 40 % of electric power and 85% of the heat generated in Russia. Analysis shows that small and medium NPP with HTGR have therefore good prospects to add to or replace these regional generation. Innovative nuclear power systems to be implemented on this basis are therefore considered as an important area of the nuclear power industry development up to the middle of the century. Based on predicted study of the power market development and demands, Russia has established Rosatom enterprise cooperation and performed pre-conceptual developmental work for commercial small MHR-100 RP prototype with modular helium reactor and several power conversion layouts as sources of various power-industrial applications.

Within the developmental work the following MHR-100 options for industrial applications were studied:

- electric power and district heat production by core thermal power conversion to electric one in direct gas turbine Brayton cycle – MHR-100 GT;
- electric power and hydrogen generation by high-temperature steam electrolysis method – MHR-100 SE;
- hydrogen generation by steam methane reforming method MHR-100 SMR;
- high-temperature heat supply to oil refinery plant – MHR-100 OR.

Development Milestones

2009	Conceptual design completed
2014	Study of necessity to adapt the design decisions to specific user requirements

Target Applications

Regional power and heat production in Russia. A single reactor unit design can be implemented in various plant configurations.

Specific Design Features

The reactor power and its design are universal for all the different power and process heat options with only the coolant parameters that are different. The reactor unit thermal power level (215 MW) was selected according to (i) the regional power industry and district heat supply needs; (ii) the manufacture needs in high- and medium-temperature heat supply for technological processes; and (iii) process capabilities of national enterprises in fabrication of RP main components including vessels.

Design Variants based on the modular MHR-100

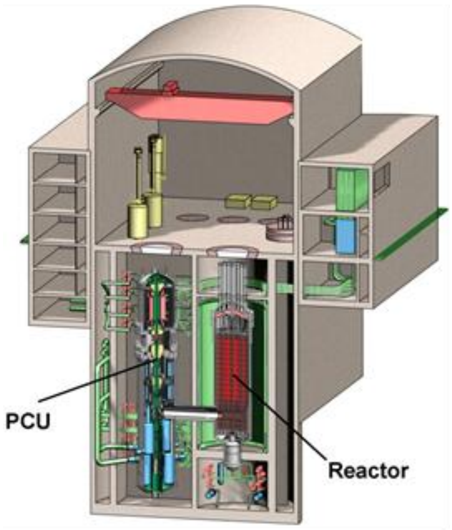
The modular reactor consists of the core with hexahedral prismatic fuel assemblies, uses helium as a coolant, and has inherent self-protection. The technical concept of studied reactor plant MHR-100 is based on:

- modular high temperature helium-cooled reactors with typical high level of inherent safety;
- fuel cycle with fuel in the form of multilayer coated fuel particles (based on UO_2), high burnup and possibility to disposal the spent fuel blocks without additional reprocessing;

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	OKBM Afrikantov
Country of Origin:	Russian Federation
Reactor Type:	Modular helium reactor
Electrical Capacity (MW(e)):	25 – 87 (depending on the configuration)
Thermal Capacity (MW(th)):	215
Expected Capacity Factor (%):	85
Design Life (years):	60
Plant Footprint (m^2):	86000
Coolant/Moderator:	Helium / Graphite
Primary Circulation:	Forced circulation
System Pressure (MPa):	4 – 5
Main Reactivity Control Mechanism:	Rod insertion
RPV Height (m):	16.82
RPV Diameter (m):	5.2
Coolant Temperature, Core Outlet (°C):	795 – 950 (see configurations)
Coolant Temperature, Core Inlet (°C):	490 – 553 (see configurations)
Integral Design:	No
Power Conversion Process:	Direct cycle Brayton, cogeneration / IHX
High-Temp Process Heat:	Yes
Low-Temp Process Heat:	Yes
Cogeneration Capability:	Yes
Design Configured for Process Heat Applications:	Yes
Passive Safety Features:	Yes
Active Safety Features:	Yes
Fuel Type/Assembly Array:	Hexagonal Prism graphite block with coated particle fuel
Fuel Active Length (m):	7.8
Number of Fuel Assemblies:	1584
Fuel Enrichment (%):	LEU < 20
Fuel Burnup (MWd/kg):	120
Fuel Cycle (months):	30
Number of Safety Trains:	2
Emergency Safety systems:	Active and passive
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	25
Distinguishing Features:	Standardized reactor design used in multiple configurations
Modules per Plant:	Depending on deployment
Estimated Construction Schedule (months):	48
Seismic Design:	8 points (MSK 64)
Predicted Core Damage Frequency:	Core damage frequency not applicable to HTGRs BDBA frequency $< 1 \times 10^{-5}$ /year Frequency of ultimate release at BDBA $< 1 \times 10^{-7}$ /year
Design Status:	Conceptual design

- high-performance high-temperature and compact heat exchangers, high-strength casings of heat-resistant steel;
- direct gas turbine cycle of power conversion with high-efficiency recuperation and intermediate coolant cooling;
- experience in high-efficiency gas turbines application in power engineering and transport;
- electromagnetic bearings used in power conversion system.

The coolant is circulated in the primary loops by the main gas circulator or by the power conversion unit (PCU) turbomachine (TM) compressors. The MHR-100 option consists of power and process parts. The power part is unified to the maximum for all options and is a power unit consisting of a reactor unit with a thermal power of 215 MW and a gas turbine PCU for power generation and (or) heat-exchange units, depending on the purpose. The process part of MHR-100 is either a process plant for hydrogen production or circuits for high-temperature heat supply to various technological processes, depending on the purpose.

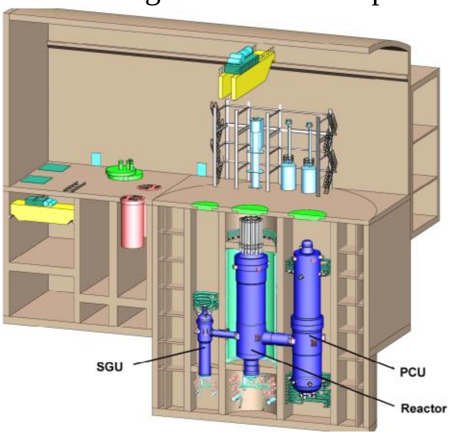


MHR-100 GT layout/options

Main parameters of MHR-100 GT

Parameter	Power mode	Cogeneration mode
Reactor heat capacity, MW	215	215
Power generation efficiency (net), %	46.1	25.4
Helium temperature at reactor inlet/outlet, °C	558/850	490/795
Low-pressure helium temperature at recuperator inlet, °C	583	595
Helium flow rate through the reactor, kg/s	139.1	134
Helium bypass flow rate from HPC outlet to recuperator outlet at high-pressure side, kg/s	-	32.2
Helium pressure at reactor inlet, MPa	4.91	4.93
Expansion ratio in turbine	2.09	1.77
Generator/TC rotation speed, rpm	3000/9000	3000/9000
PCU cooling water flow rate, kg/s	804	480
Delivery water temperature at inlet/outlet, °C	-	70/145

The unified gas turbine PCU is planned to be used in MHR-100 GT and MHR-100 SE options. Vertical oriented TM is the main feature of the PCU and consists of the turbo-compressor (TC) and generator with rotors, which have different rotation speed of 9000 rpm and 3000 rpm respectively. Electromagnetic bearings are used as the main supports. The generator is located in air medium outside the helium circuit. The PCU precooler and intercooler are arranged around TC while the recuperator is located at the top of the vessel above the hot duct axis. Waste heat from the primary circuit is removed in the PCU precooler and intercooler by the cooling water system, then in dry fan cooling towers to atmospheric air.

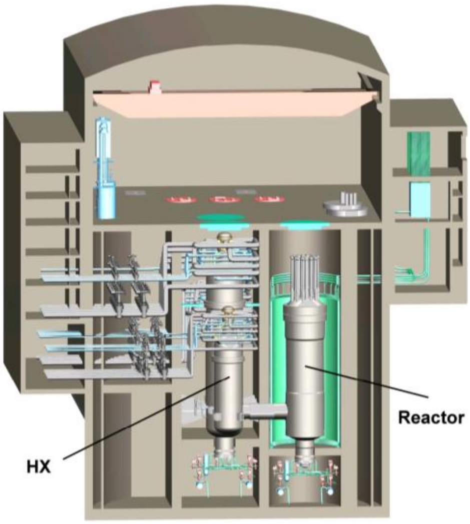


MHR-100 SE layout/options

Main parameters of MHR-100 SE

Parameter	Value
Reactor heat capacity, MW	215
Useful electric power of generator, MW	87.1
Power generation efficiency (net), %	45.7
Helium temperature at reactor inlet/outlet, °C	553/850
Helium flow rate through the reactor, kg/s	138
Helium pressure at reactor inlet, MPa	4.41
Expansion ratio in turbine	2.09
Generator/TC rotation speed, rpm	3000/9000
Helium flow rate through the turbine, kg/s	126
Helium temperature at PCU inlet/outlet, °C	850/558
SG power, MW	22.3
Helium flow rate through SG, kg/s	12.1
Helium temperature at SG inlet/outlet, °C	850/494
Steam capacity, kg/s	6.46
Steam pressure at SG outlet, MPa	4.82

Heat exchange blocks are intended to transfer heat power from the reactor to the consumer of power-technological applications. Depending on the working fluid, process type and probability of radioactivity ingress to the process product and contamination of equipment with radioactive products, two- or three-circuit RP configuration can be used. So, two circuit configurations are used in MHR-100 SE NPP for hydrogen generation and in MHR-100 SMR for steam methane reforming. Water steam is the main component of process fluid in these processes. The analysis shows that the effects of hydrogen-bearing products ingress in potential accidents with depressurization of the steam generator (SG) or high-temperature heat exchanger (HX) are reliably checked by reactor control and protection systems.

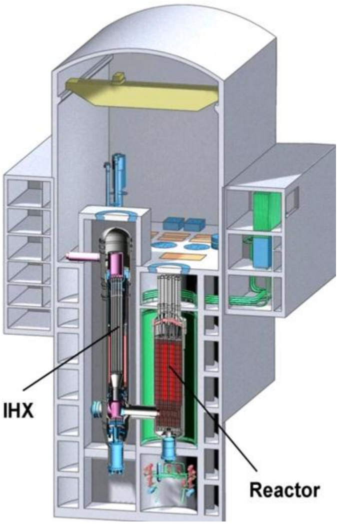


MHR-100 SMR layout/options

Main parameters of MHR-100 SMR

Parameter	Value
Reactor heat capacity, MW	215
Helium temperature at reactor inlet/outlet, °C	450/950
Helium flow rate through the reactor, kg/s	81.7
Helium pressure at reactor inlet, MPa	5.0
Steam-gas mixture pressure at HX inlet, MPa	5.3
HX-TCF 1	
HX 1 capacity MW	31.8
Helium/steam-gas mixture flow rate, kg/s	12.1/43.5
Steam-gas mixture temp. at inlet/outlet, °C	350/650
HX-TCF 2	
HX 2 capacity MW	58.5
Helium/steam-gas mixture flow rate, kg/s	22.2/60.9
Steam-gas mixture temp. at inlet/outlet, °C	350/750
HX-TCF 3	
HX 3 capacity MW	125
Helium/steam-gas mixture flow rate, kg/s	47.4/101
Steam-gas mixture temp. at inlet/outlet, °C	350/870

MHR-100 OR-based power source for heat supply to petrochemical applications and oil refinery plants has three-circuit thermal configuration. Heat from RP is transferred to the consumer via high-temperature intermediate helium-helium HX (IHX) and intermediate helium circuit and then to network circuit of petrochemical applications. This decision restricts radioactivity release to the network circuit and provides radiological purity of the process product and minimum contamination of the primary circuit with process impurities.



MHR-100 OR layout/options

Main parameters of MHR-100 OR

Parameter	Value
Reactor heat capacity, MW	215
Helium temperature at reactor inlet/outlet, °C	300/750
Helium flow rate through the reactor, kg/s	91.5
Helium pressure at reactor inlet, MPa	5.0
IHX capacity, MW	217
Primary/secondary helium flow rate through IHX, kg/s	91.5/113
Primary helium temp. at IHX inlet/outlet, °C	750/294
Secondary helium temp. at IHX inlet/outlet, °C	230/600
Secondary helium pressure at IHX inlet, MPa	5.50

Safety Features

Safety is ensured by application of passive principles of system actuation. The decay and accumulated heat is removed from the core through reactor vessel to reactor cavity cooling system and then to atmosphere by natural physical processes of heat conductivity, radiation, convection without excess of fuel safe operation limits including LOCA, in case of all active circulation systems and power sources failure. See other Russian design modular HTGR for more information. (*All figures provided courtesy of OKBM Afrikantov, with permission)*

PBMR-400 (Pebble Bed Modular Reactor SOC Ltd, South Africa)

Introduction

The Pebble Bed Modular Reactor is a High Temperature Gas Cooled Reactor based on the evolutionary design of the German HTR-Module design. The PBMR is designed in a modular fashion to allow for additional modules to be added in accordance with demand. In addition, the PBMR can be used as base-load station or load-following station, and can be configured to the size required by the community it serves.

Reactor Configuration of PBMR-400

Development Milestones

1993	The South African utility Eskom identifies PBMR as an option for new generating capacity
1995	Start of the first pre-feasibility study
1999	Design optimization: PBMR-268 with dynamic central column
2002	Design changed to PBMR-400 with fixed central column
2002	The Pebble Bed Micro Model (PBMM) demonstrated the operation of a closed, three shaft, pre- and inter-cooled Brayton cycle with a recuperator.
2004	Vertical layout of turbo machines changed to conventional single horizontal layout
2006	Commissioning of Helium Test Facility for full scale system and component tests
2006	Tests starts in the Heat Transfer Test Facility
2007	Advanced fuel coater facility commissioned
2009	Coated particles send for irradiation testing at INL
2009	Financial crises has funding implications and alternative process heat markets and designs are studied
2010	Project closure
2014	Project in care and maintenance

Target Applications

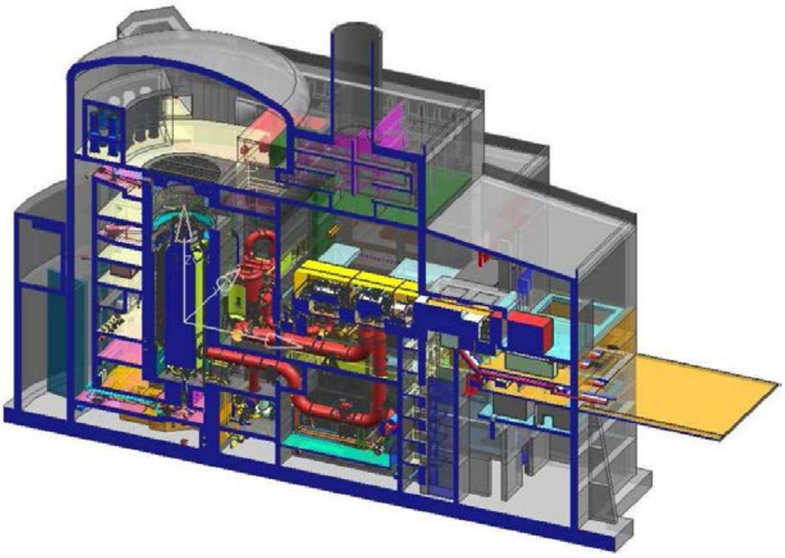
The PBMR-400 can produce electricity at high efficiency via a direct Brayton cycle employing a helium gas turbine.

The design safety targets and features means that the reactor can be deployed close to the end user since there shall be no design base or credible beyond design base event that would need anyone living near the site boundary to take shelter or be evacuated. To achieve this goal there shall be no need for engineered or moving mechanical components to ensure this target is met while the exposure to plant personnel shall also be significantly lower than today's best international practice.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	PBMR SOC Ltd
Country of Origin:	South Africa
Reactor Type:	Modular High Temperature Gas-cooled reactor
Electrical Capacity (MW(e)):	165
Thermal Capacity (MW(th)):	400
Expected Capacity Factor (%):	93
Design Life (years):	40
Plant Footprint (m^2):	4200 (main structures only)
Coolant/Moderator:	Helium / Graphite
Primary Circulation:	Forced circulation
System Pressure (MPa):	9
Main Reactivity Control Mechanism:	Negative temperature coefficient; Control rod insertion
RPV Height (m):	30
RPV Diameter (m):	6.2 (inner)
Coolant Temperature, Core Outlet (°C):	900
Coolant Temperature, Core Inlet (°C):	500
Integral Design:	No
Power Conversion Process:	Direct Brayton cycle
High-Temp Process Heat:	No, possible with different configuration
Low-Temp Process Heat:	No, possible with different configuration
Cogeneration Capability:	No, electricity only
Design Configured for Process Heat Applications:	No
Passive Safety Features:	Yes, large negative temperature coefficients, large heat capacity
Active Safety Features:	Yes, Control rod insertion with SCRAM; Turbine trip
Fuel Type/Assembly Array:	Pebble bed with coated particle fuel
Fuel Pebble Diameter (cm):	6
Number of Fuel Spheres :	~452000 in core
Fuel Enrichment (%):	9.6
Fuel Burnup (GWd/ton):	92 (average discharge)
Fuel Cycle (months):	N/A; Online / on-power refuelling
Number of Safety Trains:	N/A; The reactor cavity cooling system is not a safety system but only for investment protection. It is divided in three trains, oversized with active / passive operational modes.
Emergency Safety Systems:	Active control rod insertion
Residual Heat Removal Systems:	Active or passive reactor cavity cooling system outside reactor
Refuelling Outage (days):	N/A since online; Major outage for maintenance every 6 years
Distinguishing Features:	Inherent safety characteristics; No core melt; High efficiency; Small number of safety systems.
Modules per Plant:	1 (could also be in multi-module plant)
Estimated Construction Schedule (months):	36 months (for the nth unit)
Seismic Design:	0.4g PGA for main power system design
Predicted Large Release Frequency:	Core damage frequency not applicable to HTGRs No off-site shelter or evacuation
Design Status:	Detailed design; Test facilities demonstration; Project stopped in 2010 and design in care and maintenance.

Specific Design Features

The PBMR-400 is a high-temperature helium-cooled, graphite moderated pebble bed reactor with a multi-pass fuelling scheme. On average fuel is circulated six times through the reactor. This reduces power peaking and maximum fuel temperatures in normal operation and loss of coolant conditions. Excess reactivity is limited by continuous refuelling while adequate passive heat removal ensures an inherent safe design with no event with significant fission product release being possible. The core neutronic design results in an annular core with an outer diameter of 3.7 m and an inner diameter of 2 m shaped by the fixed central reflector. The effective cylindrical core height is 11 m. In steady state (equilibrium core) operation the fuel sphere powers (maximum 2.7 kW) and operational temperatures (<1100 ˚C) fulfil the design criteria. Adequate reactivity control and long-term cold shutdown are provided by two separate and diverse systems while the overall negative reactivity temperature coefficient is negative over the total operational range. The reactivity bound by the control rods allow xenon override for load follow between 40% and 100%

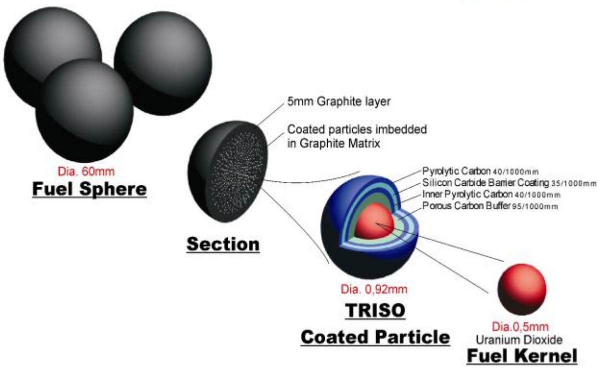


PBMR-400 building layout

Safety Features

The PBMR has a simple design basis, with passive safety features that require no human intervention and that cannot be bypassed or rendered ineffective in any way. If a fault occurs during reactor operations, the system, at worst, will come to a standstill and merely dissipate heat on a decreasing curve without any core failure or release of radioactivity to the environment. The PBMR design has the following attributes that contribute to enhanced safety:

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

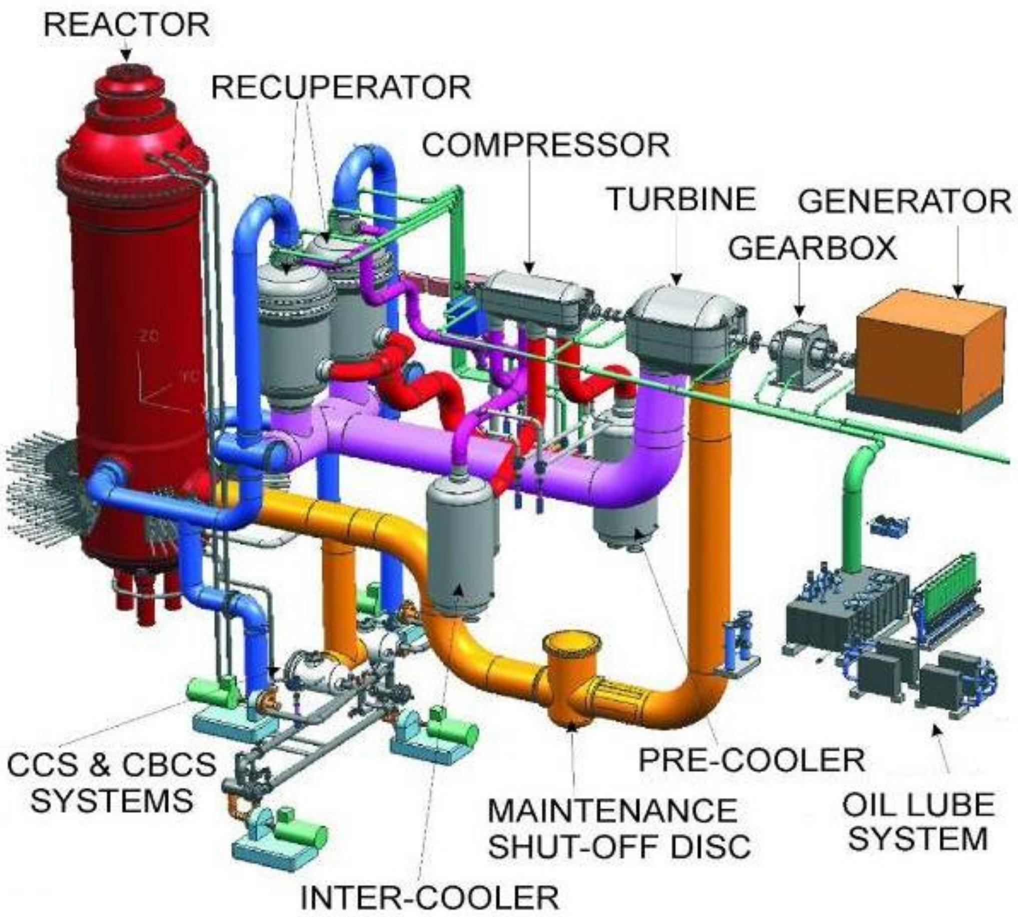


Fuel Characteristics

Coated particle pebble fuel is used. The fuel kernel (UO_2) is coated by a first porous layer of pyrocarbon, followed by a dense layer of pyrocarbon, a silicon carbide layer and an outer dense layer of pyrocarbon. About 15,000 coated particles and graphite matrix material are made into an inner fuel zone and surrounded by a 5 mm outer fuel free zone to make up the 6 cm diameter fuel sphere or pebble.

Description of the power conversion system

The Brayton power conversion with direct gas turbine is shown below.



PBMR-400 power conversion unit components and layout

Licensing and Certification Status

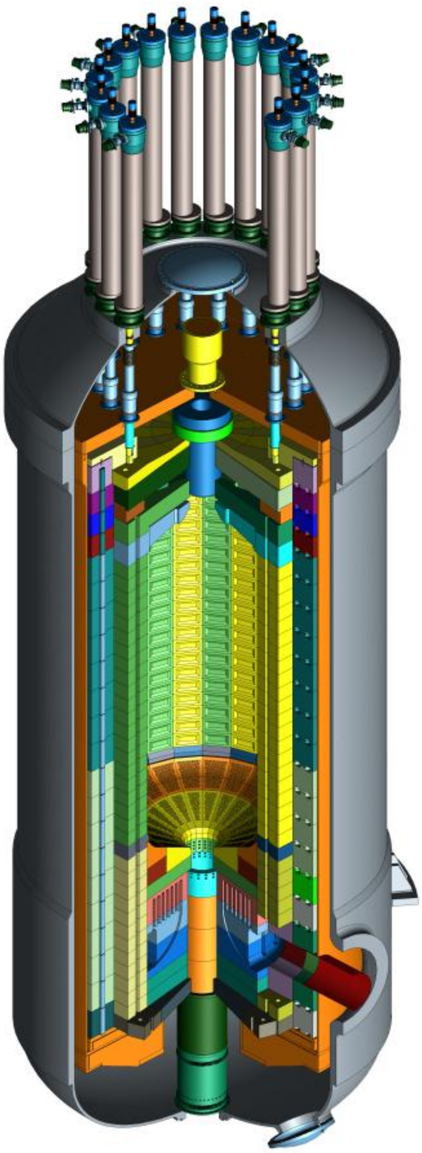
Reactor Plant Preliminary Design completed and demonstration of key technologies were underway when the project was terminated in 2010.

All figures provided courtesy of Pebble Bed Modular Reactor SOC Ltd, with permission.

HTMR-100 (STL, South Africa)

Introduction

The High Temperature Modular Reactor (HTMR-100) pebble bed high temperature gas cooled reactor, graphite moderated and cooled by forced helium. The existing design of the module is to produce high quality steam which is coupled to a steam-turbine/generator system to produce 35MW electric power. The steam can be used in a wide range of cogeneration applications. The reactor is also suitable to provide direct high temperature energy for process heat. The design of the reactor is based on proven technology and therefore no new basic technology development is needed. The size of the reactor and the fuel cycle were chosen to simplify the design and operation of the module. The approach to small intrinsic safe modular units ensures continuous production, easy road transportability, skid mounted sub systems, wider range of manufactures, fast construction and a fairly easy licensing process.



Reactor Configuration of HTMR-100
(Courtesy of STL, with permission)

Development Milestones

2012	Project started
2013	Conceptual design finished
2014	Preparation for Pre-license application

Target Applications

The Th-100 is capable of supplying electric power to large, medium and small grids, to standalone or isolated electric users as single module or multi-module plants and for medium temperature process heat applications. The HTMR-100 is a perfect fit for clients who want to progressively extend their generating capability. The unique safety characteristics make it possible to introduce and construct these plants to non-nuclear countries. First world countries that want to utilize their stock of Plutonium for peaceful applications are also markets for HTMR-100 reactors.

Specific Design Features

The reactor has good load following characteristics which is needed for stand-alone (not grid coupled) applications. The once through then out (OTTO) fuelling scheme results in a simple and cost effective fuel management system. The relative low primary loop pressure requires only a thin walled pressure vessel and thus an easier manufacturing process and a wider range of manufacturers.

The HTMR-100 plant design caters for different site and client requirements. It allows flexibility in protection against external events and flexibility in multi module configuration and power capacity.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	Steenkampskraal Thorium Limited (STL)
Country of Origin:	South Africa
Reactor Type:	High temperature Gas cooled Reactor (HTGR)
Electrical Capacity (MW(e)):	35 for single module, 140 for four module plant
Thermal Capacity (MW(th)):	100 for single module, 400 for four module plant
Expected Availability Factor (%):	> 95
Design Life (years):	40 full power years
Plant Footprint (m2):	20 000 for single module plant, 80 000 for a four module plant
Coolant/Moderator:	Helium as coolant, graphite as moderator
Primary Circulation:	Forced circulation
System Pressure (MPa):	4
Main Reactivity Control Mechanism:	Control Rods in the reflector
RPV Height (m):	15.96
RPV Diameter (m):	5.49 on flange
Coolant Temperature, Core Outlet (°C):	750
Coolant Temperature, Core Inlet (°C):	250
Integral Design:	No
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat:	Yes
Low-Temp Process Heat:	Yes
Cogeneration Capability:	Yes
Design Configured for Process Heat Applications:	Yes
Passive Safety Features:	Yes
Active Safety Features:	Yes, but no active safety engineered systems
Fuel Type:	TRISO particles in pebbles: LEU, Th/LEU, Th/HEU or Th/Pu
Fuel Size:	60 mm diameter
Number of Fuel Spheres:	~150 000 pebbles; about 120 pebbles/day throughput
Fuel Enrichment (%):	15% reactor grade Pu by mass for Th/Pu; <10% ^{235}U content (mass) for Th/LEU and Th/HEU fuel
Fuel Burnup (GWd/ton HM):	80 to 90 expected
Fuel Cycle:	Continuous online feeding
Number of Safety Trains:	No engineered safety train
Emergency Active Safety Functions:	Safety by Activation (open /close of valves and power cut only)
Residual Heat Removal Systems:	Active and Passive
Maintenance Schedule (frequency and outage(days):	Each 5 years outage of 20 days (the aim)
Distinguishing Features:	No core meltdown, modular design , fast construction, no active engineered safety system, high quality steam , reactor upgradable to very high temperate, spent fuel in highly acceptable form, about zero production of tritium thus ideal for inland sites.
Modules per Plant:	4 to 8 module packs
Estimated Construction Schedule (months):	24 months, based on continuous manufacturing and construction
Seismic Design (g):	0.3 for generic site (0.5 under consideration)
Core Damage Frequency:	Slight damage with water ingress event with design base frequency < 10^{-4} /year
Design Status:	Preparation for pre- license application

The Reactor Unit Configuration

The reactor unit consists of a steel pressure vessel, a steel core barrel, graphite reflector blocks, control rods, control rod guide tubes and drive mechanisms and in-vessel instrumentation. The vessel is designed for 4 MPa pressure and 250°C normal operational conditions in compliance with the ASME code. The Reactor unit has a weight of about 150 tons.

The graphite structure is designed to allow for differential expansion and volumetric changes due to neutron fluence. This is done to keep the stresses low and minimize primary fluid bypass and leaks. The material is standard reactor graphite.

The control and shutdown reflector rods are of different designs and are controlled independently. The structural part of the eight control rods are manufactured from high grade alloy steel and the shutdown rods from carbon carbon-fibre composites.

The primary coolant flow is from the bottom of the core barrel through risers in the outer reflector blocks to a plenum in the upper reflector. The flow through the core is from top to bottom. The heated gas is mixed in the bottom graphite structure and collected in a hot plenum. From the hot plenum the hot gas flows through the hot connecting pipe to the steam generator.

Safety Features

The Plant will be designed to withstand severe postulated conditions such as seismic, tornado and tsunami events and large airplane crashes within acceptable criteria. A conventional containment structure with associated functions like steam/gas confinement, steam condensing and hydrogen recombining is not needed for the HTMR-100 plant. Gas release during a depressurization event is released under controlled conditions, utilizing installed filter systems.

A temperature differential of approximately 700°C is maintained between the maximum allowable fuel temperature of 1 650°C and the average maximum operating temperature of the fuel elements. This allows a large temperature increase supporting a strong negative temperature coefficient for reactivity. Under these conditions reactor components do not exceed their design limits, which qualifies the function of bringing the reactor core to a sub-critical condition as a diverse shutdown function.

The low power density, the large mass of the core structures, the slender core geometry and the use of materials capable of withstanding high temperatures ensure complete passive residual heat removal capability without exceeding design limits of components.

The primary fission product barrier is the TRISO coated fuel particles, which keep the fission products contained under all postulated events, even if the second barrier (the primary pressure vessel system) and the third barrier (the filter system) collapse. A reactor with this type of fuel is classified as catastrophe free technology.

The core handles pressurized and de-pressurized loss of force cooling (PLOFC and DLOFC) scenarios successfully, without damaging fuel or reactor components.

The primary coolant is an inert gas which cannot be activated and therefor leads to cleaner environment, resulting in lower radiation dose for workers.

Fuel Characteristics and Fuel Supply Issues

The fuel element (FE) for the HTMR-100 is a 60 mm diameter sphere consisting of a spherical fuel zone of approximately 50 mm diameter, in which the TRISO-coated particles are randomly distributed in the graphitic matrix material. A fuel-free shell of graphite matrix of about 5 mm in thickness is then moulded to the fuel zone. The fuel kernel and coatings serve as an excellent fission product barrier in normal operation and accident conditions.

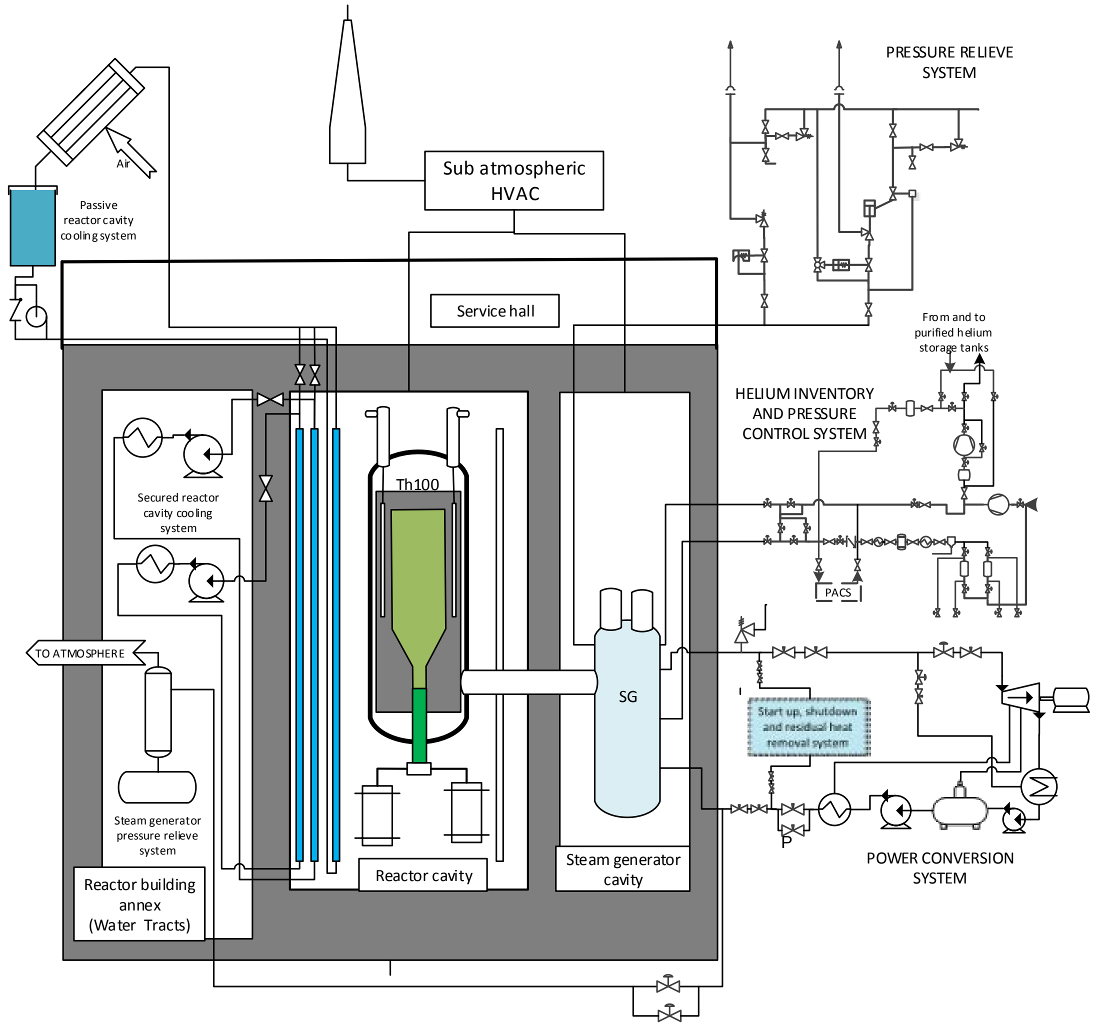
There are various types of fuel that will be used in the HTMR-100 reactor ranging from LEU UO_2 to mixtures of Th/LEU, Th/HEU and Th/Pu.

A Fuel Qualification and Test programme will be conducted on the fuel produced prior to the loading of the reactor. The approach to fuel product qualification is to ensure that Production Fuel is produced continuously from a qualified production process and facility to the same quality as the fuel developed for the German high temperature gas cooled reactors

The HTMR-100 operates on a much longer burn-up fuel cycle compared to conventional reactors. The non-proliferation characteristic of the OTTO cycle is the extended time the pebbles reside inside the core, making it impossible to divert partially burnt fuel for weapons grade uranium/plutonium extraction.

Description of the Power Conversion and Support Systems

The power conversion system is basically a conventional steam-generator unit with the condenser, supporting oil system and the control system. The Main system will be delivered in four skid mounted units namely the condenser, turbine, gearbox and electric generator. The turbine can be used in a back pressure configuration or intermediate temperature steam can be taken off for process heat applications.



Power conversion and main support systems including the primary pressure control and relieve, helium inventory control and heat removal systems

Licensing and Certification Status

Conceptual design is completed and the design is in an early stage of the basic design phase. The core, thermo-hydraulic and other analysis are needed to optimize the processes and to perform the safety analysis. The aim is to apply for a typical pre-license or pre-certification within 18 months (by end of 2015).

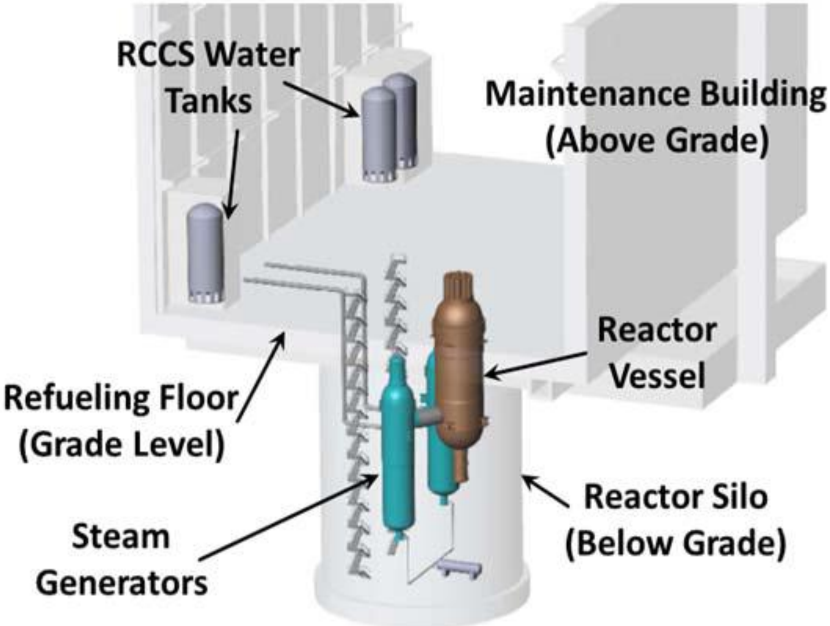
STL has recently entered cooperation with a number of organizations to form HTMR Limited, an international company devoted to the development of small, modular HTGR, with its headquarters located in UK. *(All figures provided courtesy of STL, with permission).*

SC-HTGR (AREVA, USA)

Introduction

AREVA's Steam Cycle High Temperature Gas-Cooled Reactor (SC-HTGR) is a small, modular, graphite-moderated, and helium-cooled, high temperature reactor with a nominal thermal power of 625 MW(th). Its intrinsic safety characteristics make it uniquely qualified to be collocated with a commercial industrial facility and to be used to provide energy in the form of process steam and/or electricity to that facility. This concept couples proven gas reactor technology with a modular deployment strategy to provide an effective match with a range of potential end user industrial processes.

The SC-HTGR concept builds on the experience of past HTGR projects, as well as development and design advances that have taken place in recent years. The SC-HTGR uses a prismatic block type reactor core based on AREVA's ANTARES concept, which is sized to take maximum advantage of the passive heat removal capability of modular HTRs.



Single SC-HTGR Reactor Module
(Courtesy of AREVA, with permission)

Development Milestones

2009	Project Started
2011	Concept Definition Completed
2012	NGNP Industry Alliance Selected the technology for Commercialization
2012	NGNP Industry Alliance Completed an initial Business Plan for SC-HTGR Commercialization
2014	NGNP Industry Alliance Updated the SC-HTGR Commercialization Plan
2014	Preparation for Pre-Licensing Application Started

Target Applications

The SC-HTGR steam cycle-based concept is extremely flexible. Since high pressure steam is one of the most versatile heat transport mediums, a single basic reactor module configuration designed to produce high temperature steam is capable of serving a wide variety of near-term markets. As a result, the standard SC-HTGR is well suited to supply a wide variety of process heat facilities. The steam system equipment can be configured in a variety of ways depending on the specific needs of the facility for high temperature steam, low temperature steam, and electricity.

Perhaps more importantly the steam cycle is well suited to cogeneration of electricity and process heat. Using the conventional Rankine cycle with high temperature steam, a net efficiency of about 43% can be achieved in the full electricity generation mode. This makes the concept an attractive option in markets with limited grids and markets requiring incremental capacity addition.

The steam cycle plant also has good load following characteristics. Reactor module power level and steam production can be increased or decreased relatively easily. Systems can also shift energy between electricity generation and heat supply dynamically as load conditions vary, all while keeping reactor power constant. This provides the maximum utilization of the HTR nuclear heat source.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	AREVA
Country of Origin:	USA
Reactor Type:	Prismatic Block HTGR
Electrical Capacity (MW(e)):	272
Thermal Capacity (MW(th)):	625
Design Life (years):	60
Coolant/Moderator:	Helium/Graphite
Primary Circulation:	Forced circulation
Primary Coolant Pressure (MPa):	6
Coolant Temperature, Core Outlet (°C):	750
Coolant Temperature, Core Inlet (°C):	325
Vessel Material:	SA508/533 (typical PWR vessel material)
Secondary Coolant:	Water/Steam
Power Conversion Process:	Rankine Cycle
Steam Generator Power (MW(th)):	315 (each)
Main Circulator Power (MW(e)):	4 (each)
Main Steam Temperature (°C):	566
Main Steam Pressure (MPa):	16.7
Net Electric Output in All Electric Mode (MW(e)):	272
High-Temp Process Heat:	Yes
Low-Temp Process Heat:	Yes
Cogeneration Capability:	Yes
Design Configured for Process Heat Applications:	Yes
Passive Safety Features:	Yes
Active Safety Features:	No
Fuel Type/Assembly Array:	UCO TRISO Particle fuel in hexagonal graphite blocks
Fuel Block Height (m):	0.8
Number of Fuel Assemblies:	1020 (102 columns of 10 blocks each)
Fuel Enrichment (%):	< 20
Fuel Burnup (FIMA) (%):	< 17
Fuel Cycle (months):	18 – 24
Emergency Safety Systems:	Passive
Residual Heat Removal Systems:	Active/Passive
Refuelling Outage (days):	25
Modules per Plant:	Variable (Reference design is four 625 MWth modules)

Specific Design Features

The SC-HTGR uses a two-loop modular steam supply system. Each module consists of one reactor coupled to two helical coil steam generators. The steam generators are configured in parallel, each with a dedicated main circulator. A steel vessel system using conventional vessel material houses the entire primary circuit. The entire inner vessel surface is bathed in cool reactor inlet gas. Electric, variable speed circulators, using submerged motors with active magnetic bearings, provide the primary coolant flow. Each reactor module is located in a separate, fully embedded, below grade reactor building. This provides structural design advantages and superior protection from external hazards.

Safety Features

The primary safety objective of the SC-HTGR design is to limit the dose from accidental releases so that the US EPA Protective Action Guides are met at an exclusion area boundary only a few hundred meters from the reactor. To achieve this safety objective, the design uses the high temperature capabilities of TRISO-coated fuel particles, graphite moderator, and helium coolant, along with the passive heat removal capability of a low power-density core and an un-insulated steel reactor vessel.

The primary radionuclide retention barrier in the SC-HTGR consists of the three ceramic coating layers surrounding the fuel kernel that forms a fuel particle. The coating system constitutes a miniature pressure vessel around each kernel that has been engineered to withstand extremely high temperatures without losing its ability to retain radionuclides even under accident conditions.

The high temperature capabilities of the massive graphite reactor core structural components complement the fuel's high temperature capability. The high heat capacity and low power density of the core result in very slow and predictable temperature transients even without cooling. Helium, the reactor coolant and heat transport medium, is chemically inert and neutronically transparent, Helium will not change phase during normal operation or accidents.

The SC-HTGR is designed to passively remove decay heat from the core regardless of whether the primary coolant is present. The concrete walls surrounding the reactor vessel are covered by the Reactor Cavity Cooling System panels, which provide natural circulation cooling during both normal operation and accidents, so there is no need for the system to change modes or configuration in the event of an accident. Moreover, the thermal characteristics of the reactor are such that even if the RCCS were to fail during an accident, the safety consequences would still be acceptable.

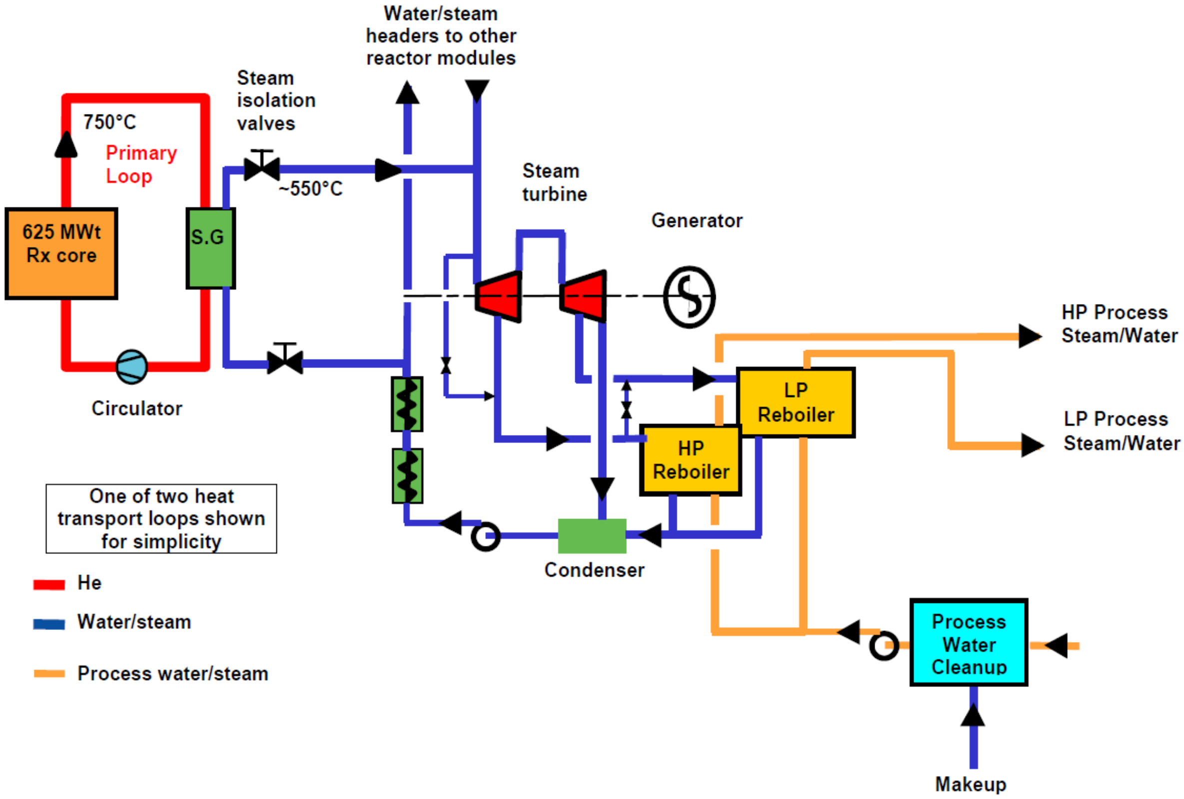
The large negative temperature coefficient of the modular SC-HTGR, along with its large thermal margins, provide for an inherent shutdown capability to deal with failures to scram the reactor. Gravity-driven and diverse reactivity control systems provide further confidence of the ability to shut down the reactor. No powered safety-related systems and no operator actions are required to respond to any of the accident scenarios that have been postulated for the modular HTGRs throughout their licensing history.

SC-HTGR Plant Arrangement

The modular design of the SC-HTGR allows multiple reactor modules to be grouped together on a single plant sight. A typical plant layout might have four reactor modules; however, the specific number of modules in an actual plant will depend on the nature of the application and the customer's needs.

Reactor modules share auxiliary and supporting systems during normal operation, but safety systems are independent. Each reactor module has independent control and protection systems. A common supervisory control system coordinates the interface between the reactor modules, process steam demand(s), and electricity generation. The supervisory control system allocates load demand between individual reactor modules and provides inputs to the independent module control systems accordingly. The power conversion system portion of each plant will be customized for the application of interest. In a typical plant providing steam for process heat and/or cogeneration, steam from multiple reactor modules is supplied to a common header which supplies one or more turbines or process loads.

While the energy delivery system for an SC-HTGR plant will normally be customized for the specific needs of each customer, the actual reactor modules are completely standardized. This is the key to successful commercialization of the SC-HTGR. The whole NSSS, including the reactor, the steam generators, the circulators, and the surrounding vessel system is identical for each module. The reactor modules are each housed in separate reactor buildings which are also identical. And primary supporting services such as fuel handling systems, reactor module control and protection systems, and primary auxiliary systems are the same for each reactor installation.

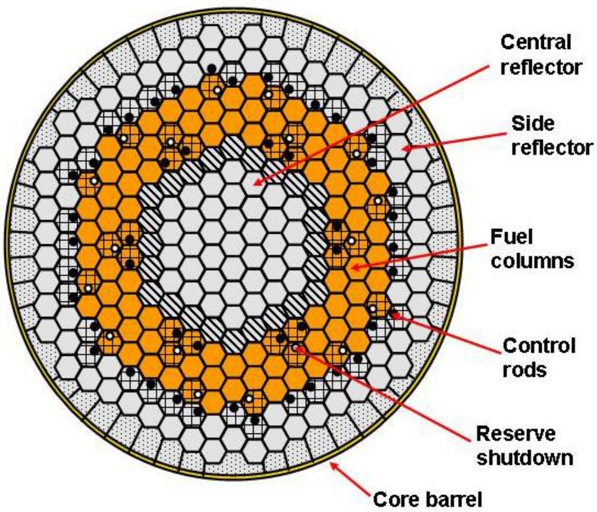


Typical Cogeneration Plant Configuration (Courtesy of AREVA, with permission)

Fuel Characteristics and Fuel Supply Issues

The heart of the AREVA steam cycle concept is the prismatic block reactor. It is a 102 column annular core. This geometry provides good radial heat conduction to maximize the benefits of passive decay heat removal. The reactor uses TRISO coated particle fuel. At the center of each fuel particle is a kernel of uranium oxycarbide (UCO). The coated fuel particles are embedded in cylindrical compacts of graphite matrix material slightly over 1 cm in diameter. The compacts are loaded into graphite fuel blocks. The nominal power distribution is controlled in the core design using a combination of particle packing fraction, burnable poison, and enrichment as appropriate for each specific fuel cycle. The enrichment is less than 20 percent.

The nominal power distribution is controlled in the core design using a combination of particle packing fraction, burnable poison, and enrichment as appropriate for each specific fuel cycle. The enrichment is less than 20 percent. Normal reactor control uses control rods in channels at the edge of the core. An alternate reserve shutdown system is also available. The reserve shutdown system drops absorber material into separate channels in the core interior when activated. Either the control rods or the reserve shutdown system can shut the reactor down. And, if neither system functions, negative temperature reactivity feedback will shut the reactor down with only a slight temperature increase.



SC-HTGR Core Configuration (Courtesy of AREVA, with permission)

Licencing and Certification Status

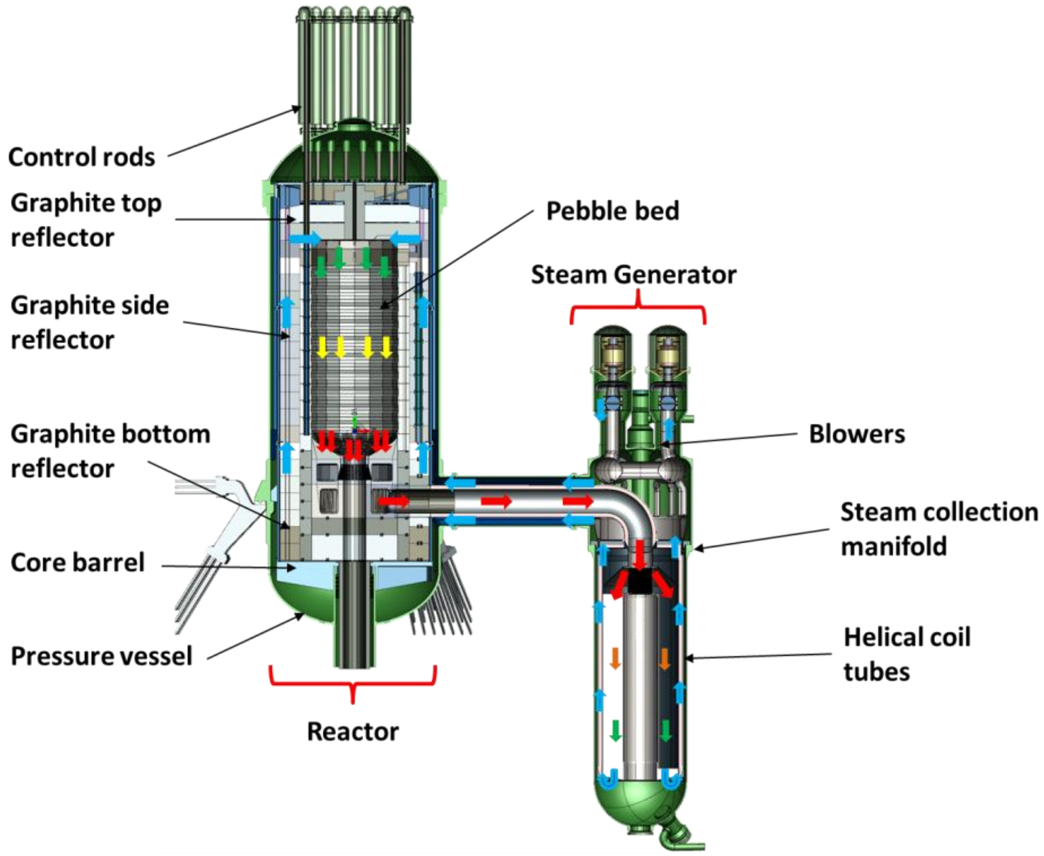
Conceptual Design Completed and preparing for Pre-Licensing Application.

Xe-100 (X-energy, USA)

Introduction

The Xe-100 is a small-sized pebble bed high temperature gas-cooled reactor with continuous thermal rating of 100 MW. It features a continuous fuelling regime with low enriched fuel spheres of about 10 wt% entering the top of the reactor going once through the core to achieve a final average burnup of 80 000 MWd/tHM after a single passage. The relatively high burnup causes the bred fissile Pu to be utilized in-situ by about 90%, thus rendering the spent fuel well depleted. Furthermore, the total used fuel inventory will be stored on-site in a designated interim storage facility for the life of the plant.

A major aim of the design is to improve the economics through system simplification, component modularization, reduction of construction time and high plant availability brought about by continuous fuelling.



Xe-100 Reactor System Configuration (Courtesy of X-energy, with permission)

Development Milestones

2013	Conceptual Design Development
2015	Base Design and Pre-application with USNRC
2017	Application with USNRC
2021	First operation and commercialization

Target Applications

The Xe-100 is intended for electricity production suitable for small or isolated grids. It can also provide super-heated steam for cogeneration, petro-chemical processes, etc. The Xe-100 also provides a scalable platform to increase total power generation from a single site by adding additional reactor modules. Site configurations can consist of 1 to 8 reactor modules with a small operational staff.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer:	X-energy LLC.
Country of Origin:	USA
Reactor Type:	Modular High Temperature Gas-cooled Reactor (HTGR)
Electrical Capacity (MW(e)):	35
Thermal Capacity (MW(th)):	100
Expected Availability Factor (%):	> 95
Design Life (years):	40 with potential life extensions
Plant Footprint (m^2):	100 000 for (4 Reactor modules with one turbine); 160 000 (8 Reactor modules with two turbines)
Coolant/Moderator:	He / Graphite
Primary Circulation:	Forced confection; He flow Top → Bottom through core
System Pressure (MPa):	4
Reactivity Control and Shutdown System (RCS + RSS):	Bank 1 (9 Rods) + Bank 2 (9 Rods)
RCSS Location From the Core Face in Side Reflector (cm):	8
RCS Insertion Depth (normal operation) (m):	1.335
RPV Height (m):	15
RPV Diameter (m):	5.15
Coolant Temperature, Core Outlet (°C):	750
Coolant Temperature, Core Inlet (°C):	260
Integral Design:	No
Power Conversion Process:	Indirect Rankine Cycle
High-Temp Process Heat	Yes, possible
Low-Temp Process Heat:	Yes, possible
Cogeneration Capability:	Yes, possible
Design Configured for Process Heat Applications:	Yes, possible
Passive Safety Features:	Yes
Active Safety Features:	Yes (Defence in depth & investment protection systems)
Fuel Type:	Pebbles
Core Volume (m^3):	27.7
Fuel Enrichment (wt%):	10.61
Fuel Residence time (d):	1149
Burnup (MWd/kg_{HM}):	79.7
Fuel Cycle:	On-line refuelling
Residual Heat Removal Systems:	Passive and Active systems
Power Peaking (Q_{max}/Q_{avg}):	2.40
Max. Power per FS (kW/sphere):	1.60
Distinguishing Features:	Online refuelling, core cannot melt and fuel damage minimized by design, independent fission product barriers, potential for advanced fuel cycles
Modules per Plant	2 – 8
Estimated Construction Schedule (months):	24 – 36
Seismic Design (g):	0.15 operating limit & 0.3 safe shutdown
Core Damage Frequency:	No core melt possible
T_{max} of Fuel in Normal Operation (°C):	834
T_{max} of Fuel in DLOFC (DH = 1.1) (after h) (°C):	1617 (15)
Design Status:	Conceptual design development

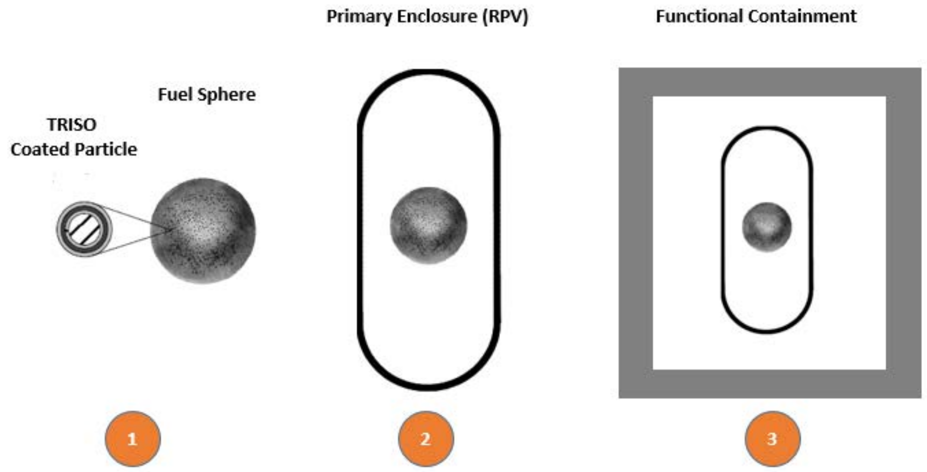
Specific Design Features

The Xe-100 pebble bed HTR is a simplified design that utilizes well-proven materials and components to expedite the licensing process. It is therefore utilizing a low enriched fuel cycle, with TRISO coatings embedding UCO fuel kernels. Due to its seemingly better performance parameters this fuel option allows for a dramatic increase in power rating or in burn-up. The decision is pending finalization of the DOE funded UCO fuel qualification program.

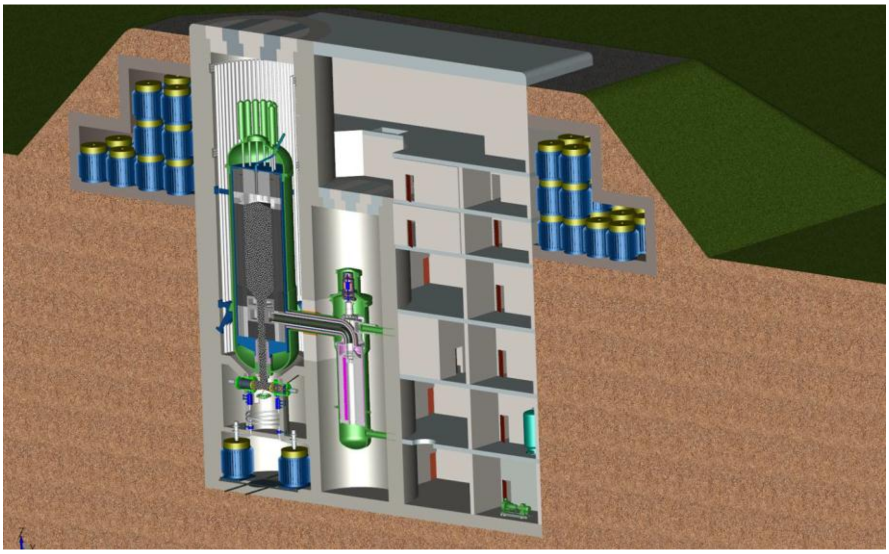
Furthermore, the Xe-100 module is designed for a flexible advanced fuel cycle based on either uranium or thorium. The double heterogeneous particulate design enables the possibility to operate under either fuel cycle. Not only does this afford the Xe-100 the potential entertain a fuel cycle that will produce virtually no reactor grade plutonium and minor actinides, but also yield the potential of utilizing the Xe-100 as Pu-incinerator coming from excess weapons' programs or from a reprocessing process, such as that used by the French or Japanese.

Safety Features

The intrinsic safety characteristic of the plant is guaranteed by a relatively low power density of about 3.7 MW/m^3, high thermal inertia of the graphitic internals layout and a strong negative temperature coefficient of reactivity over the total operational regime of the reactor. This feature is further enhanced via a unique system of *independent fission product barriers*. The effects of possible external impacts (such as aircraft or adverse weather) are also mitigated by the proposed, partially below ground, module layout.

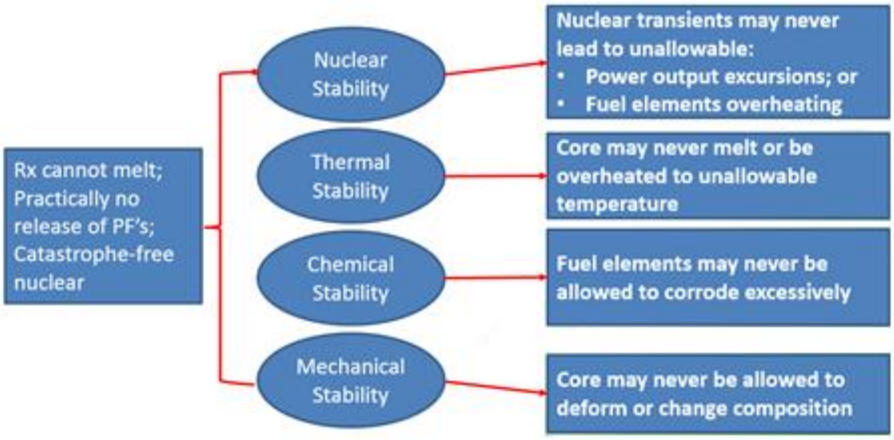


Concept of Independent Fission Product Barriers



Conceptual Reactor Module Design (Courtesy of X-energy, with permission)

The Xe-100 design is based on the 4 principles of stability depicted below. The fact that the maximum fuel temperatures will remain below proven safety performance limits of TRISO fuel means that no core melt will be physically possible and no significant core damage is possible that would necessitate the evacuation of the general public or cause significant environmental damage under normal operation and any design basis events



Stability Design: 4 Principles

Fuel Characteristics and Fuel Supply Issues

In reactor design a Cosine-shaped axial flux profile is aimed for. This will yield an optimal power profile as the spatial neutron reaction rate, RR [$cm^{-3} \cdot s^{-1}$]:

$$RR(\vec{r}) = \int_{(E)} \emptyset(\vec{r}, E) \cdot \sigma(E) \cdot N \cdot dE$$

with: $\emptyset(\vec{r}, E) \cdot dE$ = Integrated neutron flux distribution, together with the reaction rate, determined by neutron diffusion.

Every time that a fuel sphere reaches the bottom of the core the level of burn-up is determined via active measurement of targeted isotopes in a gamma spectrometer through the build-up of a preselected long-lived isotope, such as ^{137}Cs.

The OTTO (Once-Through-Then-Out) fuel cycle is the simplest that one can design for, even though it delivers a more asymmetric axial flux profile. In this arrangement the burnup complex can, however, be eliminated as standard equipment, as well as a complicated return system of used fuel from the bottom of the core back to the top. As this implies the replacement of a complex fuelling/de-fuelling system with a relatively simple one it makes good sense to consider deploying the system as such.

As noted above the fuel used will be TRISO coated UCO fuel with kernels of slightly smaller diameter (425 μm) than the usual UO_2 fuel (500 μm). The rest of the fuel materials data will remain similar to the German UO_2 data. The optimized moderation ratio (N_C/N_A) has led to a heavy metal loading of around 9 g/pebble. This would enable the Xe-100 under worst case water ingress scenarios to be able to shut down the reactor with its RCSS.

It is also noted above that a power peaking of 2.4 is observed. Furthermore, the maximum power rating of the pebbles remain well within the operational envelope of performance experimentally determined.

Fuel cycle costs have also been estimated according to the Present Worth method. Subject to a complex series of economical parameters and assumptions figures around 7.24 Mills/kWh(e) are estimated.

Licensing and Certification Status

Conceptual design development and preparation for pre-licensing application.

Annex I

SUMMARY OF SMR DESIGN STATUS

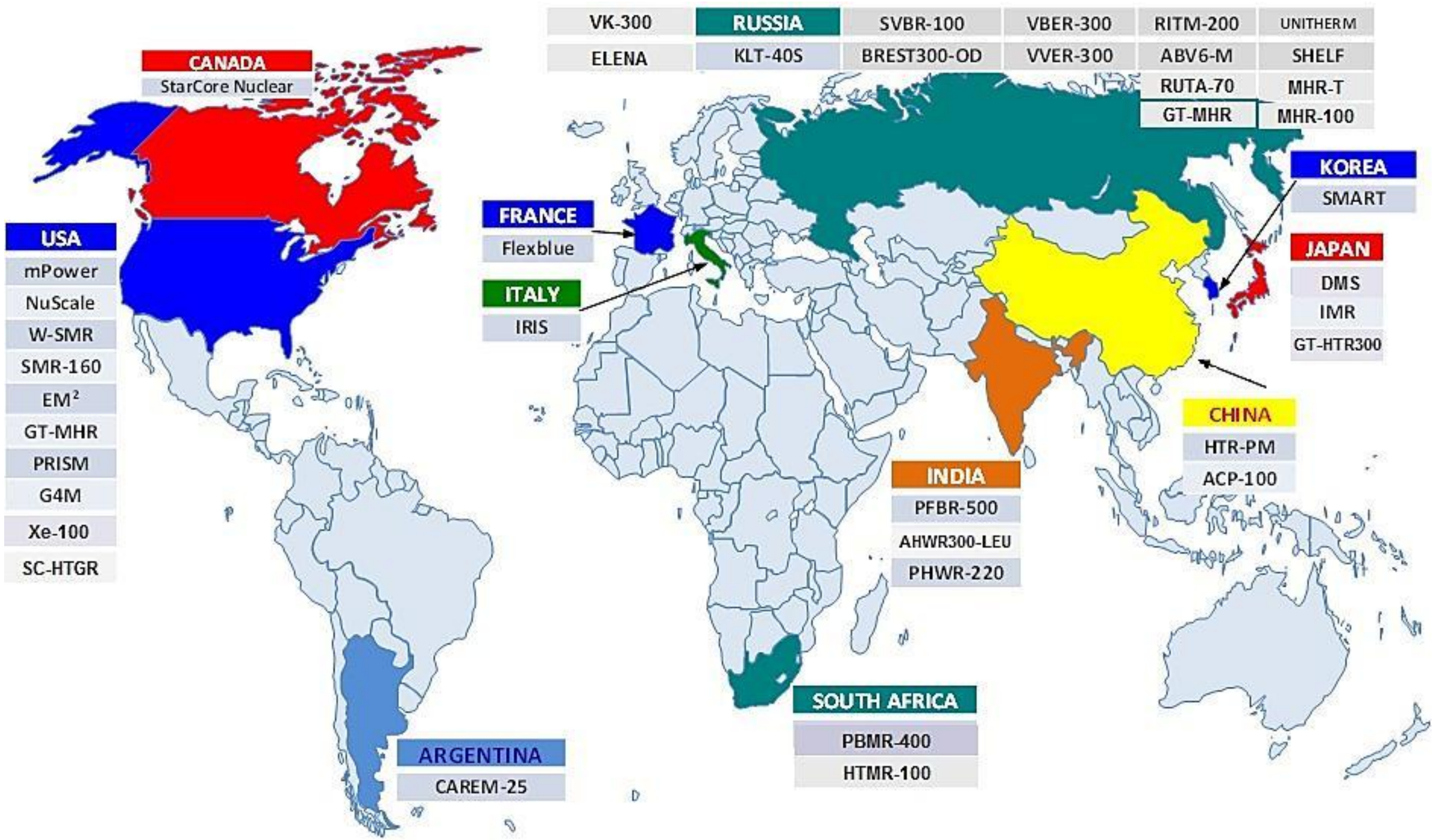
Reactor design	Reactor type	Designer, country	Capacity(MW(e))/ configuration	Design status
WATER COOLED REACTORS				
CAREM-25	Integral pressurized water reactor	CNEA, Argentina	27	Under construction
ACP-100	Integral pressurized water reactor	CNNC (NPIC/CNPE), China	100	Detailed design
Flexblue	Subsea pressurized water reactor	DCNS, France	160	Conceptual design
AHWR300-LEU	Pressure tube type heavy water moderated reactor	BARC, India	304	Basic design
IRIS	Integral pressurized water reactor	IRIS, International Consortium	335	Basic design
DMS	Boiling water reactor	Hitachi-GE Nuclear Energy, Japan	300	Basic design
IMR	Integral modular water reactor	Mitsubishi Heavy Industries, Japan	350	Conceptual design completed
SMART	Integral pressurized water reactor	KAERI, Republic of Korea	100	Licensed/Certified
KLT-40S	Pressurized water reactor	OKBM Afrikantov, Russian Federation	35 × 2 modules barge mounted	Under construction, planned commercial start 2016-2017
VBER-300	Integral pressurized water reactor	OKBM Afrikantov, Russian Federation	325	Licensing stage
ABV-6M	Pressurized water reactor	OKBM Afrikantov, Russian Federation	6 × 2 modules, barge mounted, land based	Detailed design

RITM-200	Integral pressurized water reactor	OKBM Afrikantov, Russian Federation	50	Under construction, planned commercial start 2017
VVER-300	Water-cooled water-moderated power reactor	OKB Gidropress, Russian Federation	300	Conceptual design
VK-300	Simplified boiling water reactor	RDIPE, Research and Development Institute of Power Engineering, Russian Federation	250	Detailed design of Reactor and Cogeneration Plant Standard Design
UNITHERM	Pressurized water reactor	RDIPE, Research and Development Institute of Power Engineering, Russian Federation	6.6	Conceptual design
RUTA-70	Pressurized water reactor	RDIPE, Research and Development Institute of Power Engineering, IPPE, Russian Federation	70	Conceptual design
SHELF	Pressurized water reactor	RDIPE, Research and Development Institute of Power Engineering, Russian Federation	6	Conceptual design
ELENA	Pressurized water reactor	Research Russian Centre "Kurchatov Institute", Russian Federation	0.068	Conceptual design
mPower	Integral pressurized water reactor	B&W Generation mPower, USA	180 × 2 modules	Basic design
NuScale	Integral pressurized water reactor	NuScale Power LLC., USA	45 × 12 modules	Basic design
Westinghouse SMR	Integral pressurized water reactor	Westinghouse Electric Company LLC, USA	>225	Preliminary design completed
SMR-160	Pressurized water reactor	Holtec International, USA	160	Conceptual design

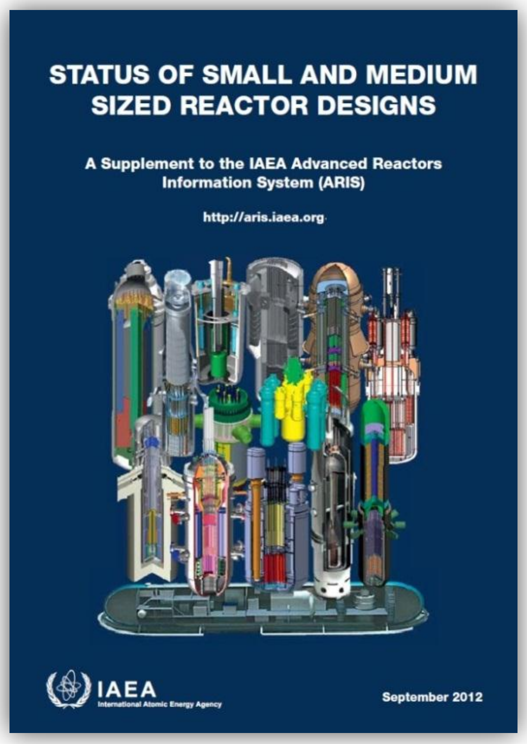
HIGH TEMPERATURE GAS COOLED REACTORS				
HTR-PM	Pebble Bed HTGR	Tsinghua University, China	211	Under construction
GT-HTR300	Prismatic Block HTGR	Japan Atomic Energy Agency, Japan	100 – 300	Basic design
GT-MHR	Prismatic Block HTGR	OKBM Afrikantov, Russian Federation	285	Conceptual design completed
MHR-T reactor/Hydrogen production complex	Prismatic Block HTGR	OKBM Afrikantov, Russian Federation	4 x 205.5 Hydrogen production	Conceptual design
MHR-100	Prismatic Block HTGR	OKBM Afrikantov, Russian Federation	25 – 87 cogeneration	Conceptual design
PBMR-400	Pebble Bed HTGR	Pebble Bed Modular Reactor SOC Ltd, South Africa	165	Detailed design
HTMR-100	Pebble Bed HTGR	Steenkampskraal Thorium Limited (STL), South Africa	35 per module (140 for 4 module plant)	Conceptual design, preparation for pre-license application
SC-HTGR	Prismatic Block HTGR	AREVA, USA	272	Conceptual design
Xe-100	Pebble Bed HTGR	X-energy, USA	35	Conceptual design

Annex II

MAP OF GLOBAL SMR TECHNOLOGY DEVELOPMENT



Annex III
BIBLIOGRAPHY



- Contained status, design description and main features of 32 selected SMR designs;
- Sorted by type/coolant: iPWR, PHWR, GCR, and LMFR;
- Sorted by Country of Origin;
- Included: CAREM (Argentina), FBNR (Brazil), CNP-300 (China), Flexblue (France), IMR (Japan), SMART (Republic of Korea), ABV-6M (Russian Federation), SHELF (Russian Federation), RITM-200 (Russian Federation), VK-300 (Russian Federation), VBER-300 (Russian Federation), WWER-300 (Russian Federation), KLT-40S (Russian Federation), UNITHERM (Russian Federation), IRIS (International Consortium), mPower (USA), NoScale (USA), Westinghouse SMR (USA), EC6 (Canada), PHWR-220 (India), AHWR300-LEU (India), HTR-PM (China), PBMR (South Africa), GT-MHR (USA), EM^2 (USA), CEFR (China), 4S (Japan), PFBR-500 (India), BREST-OD-300 (Russian Federation), SVBR-100 (Russian Federation), PRISM (USA), G4M (USA).

- Contained status, design description and main features of 22 selected fast reactor designs;
- Sorted by type/coolant: SFR, GFR, and HLMC, MSFR;
- Sorted by Country of Origin;
- Included: CFR-600 (China), ASTRID (France), FBR-1&2 (India), 4S (Japan), JSFR (Japan), PGSFR (Republic of Korea), BN-1200 (Russian Federation), MBIR (Russian Federation), PRISM (USA), TWR-P (USA), MYRRHA (Belgium), CLEAR-I (China), ALFRED (Europe/Italy), ELFR (Europe/Italy), PEACER (Republic of Korea), BREST-OD-300 (Russian Federation), SVBR-100 (Russian Federation), ELECTRA (Sweden), G4M (USA), ALLEGRO (Europe), EM^2 (USA), MSFR (France).

Annex IV

ACRONYMS

AC	Alternating Current
ADS	Automatic Depressurization System
ARIS	Advanced Reactor Information System
BCR	Back-up Control Room
BDBA	Beyond Design Basis Accident
BOP	Balance of Plant
BWR	Boiling Water Reactor
CCWS	Component Cooling Water System
CEDM	Control Element Drive Mechanism
CES	Containment Enclosure Structure
CMT	Core Make-up Tank
CPS	Control and Protection System
CRDM	Control Rod Drive Mechanism
CS	Containment Structure
CSS	Control Safety System
CTS	Chemical Technological Sector
CV	Containment Vessel
CVCS	Chemical and Volume Control System
DAS	Diverse Actuation System
DBA	Design Basis Accident
DC	Direct Current
DID	Defence in Depth
DLOFC	Depressurized Loss of Forced Cooling
DVI	Direct Vessel Injection
ECCS	Emergency Core Cooling System
ECDS	Emergency Cooling Down System
ECT	Emergency Cooldown Tank
EDG	Emergency Diesel Generator
EHRS	Emergency Heat Removal System
EPZ	Emergency Planning Zone
ESWS	Essential Service Water System
FA	Fuel Assembly
FE	Fuel Element
FPU	Floating Power Unit
FSAR	Final Safety Analysis Report
FSS	Free Surface Separation
GDCS	Gravity Driven Cooling System
GDWP	Gravity Driven Water Pool
HEU	High Enriched Uranium
HHTS	Hybrid Heat Transport System
HPCF	High Pressure Core Flooder
HTGR	High Temperature Gas-cooled Reactor
HTR	High Temperature Reactor
HX	Heat Exchanger
IC	Isolation Condenser
I&C	Instrumentation and Control
LEU	Low Enriched Uranium
LOCA	Loss of Coolant Accident
LOOP	Loss of Offsite Power

LPFL	Low Pressure Core Flooder
LWR	Light Water Reactor
MCR	Main Control Room
MHT	Main Heat Transport
MOX	Mixed Oxide
MSA	Moisture Separator Reheater
MW(e)	Mega Watt electric
MW(th)	Mega Watt thermal
NDHP	Nuclear District Heating Plant
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
OBE	Operating Basis Earthquake
OCP	Outside Containment Pool
OTTO	Once Through Then Out
PC	Primary Containment
PCT	Peak Cladding Temperature
PCU	Power Conversion Unit
PCV	Primary Containment Vessel
PCCS	Passive Containment Cooling System
PLOFC	Pressurized Loss of Forced Cooling
PLS	Plant Control System
PMS	Protection and safety Monitoring System
PORV	Power-Operated Relieve Valve
PRHRS	Passive Residual Heat Removal System
PSAR	Preliminary Safety Analysis Report
PSIS	Passive Safety Injection System
PWR	Pressurized Water Reactor
RCIC	Reactor Core Isolation Cooling
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RCSS	Reactivity Control and Shutdown System
RFA	Robust Fuel Assembly
RHRS	Residual Heat Removal System
RP	Reactor Plant
RPV	Reactor Pressure Vessel
RV	Reactor Vessel
SBO	Station Black-Out
SG	Steam Generator
SSE	Safe Shutdown Earthquake
TC	Turbo Compressor
TEG	Thermoelectric Generator
TEU	Thermoelectric Unit
TM	Turbo Machine
TRISO	Triple Coated Isotropic
UCO	Uranium Oxy Carbide
UHS	Ultimate Heat Sink
WDS	Waste Disposal System
WPu	Weapon-Grade Plutonium
WWER	Water Moderated Power Reactor

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