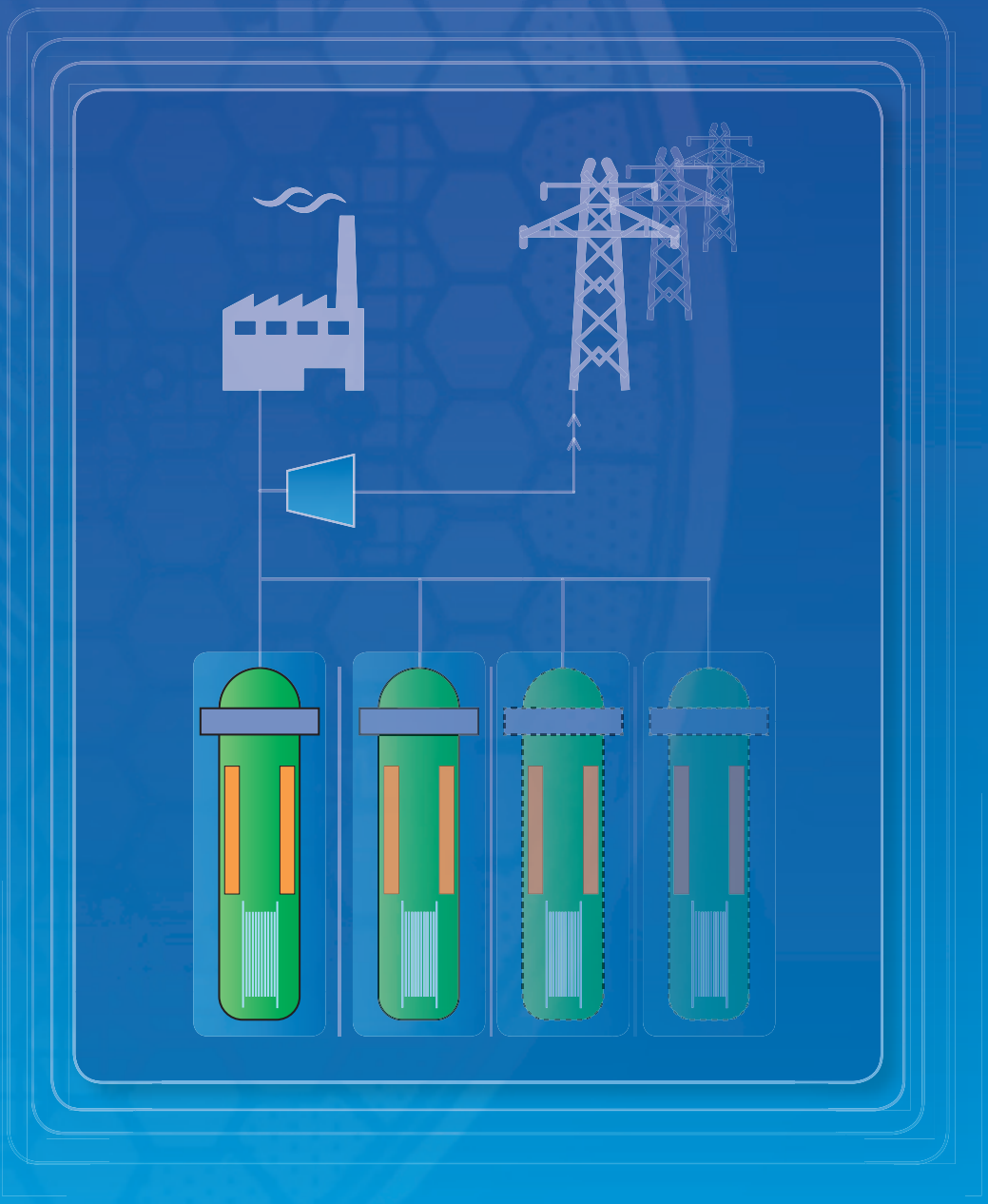
Advances in Small Modular Reactor Technology Developments

A Supplement to:
IAEA Advanced Reactors Information System (ARIS)
2016 Edition



IAEA
International Atomic Energy Agency

ADVANCES IN SMALL MODULAR REACTOR TECHNOLOGY DEVELOPMENTS

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http://aris.iaea.org

Printed by the IAEA in Austria
August 2016

FOREWORD

The IAEA Department of Nuclear Energy continues to facilitate efforts of Member States in the development and deployment of small modular reactors (SMRs), recognizing their potential as a viable solution to meet energy supply security, both in newcomer and expanding countries interested in SMRs. In this regards, balanced and objective information to all Member States on technology status and development trends for advanced reactor lines and their applications are collected, assessed and provided through publication of status reports and other technical documents.

Member States, both those launching their nuclear power programme and those with an existing nuclear power programme, keep expressing their interest in information about advanced SMR designs and concepts, as well as current development trends. The IAEA Division of Nuclear Power, which has been facilitating Member States in addressing common technologies and issues for SMRs and the related fuel-cycle, plays a prominent role in convening international scientific forums and technical cooperation in this field for the interested Member States. The activities on SMRs are further supported by specific activities on advance fast and high temperature gas cooled reactor technology development.

The driving forces in the development of SMRs are their specific characteristics. They can be deployed incrementally to closely match increasing energy demand resulting in a moderate financial commitment for countries or regions with smaller electricity grids. SMRs show the promise of significant cost reduction through modularization and factory construction which should further improve the construction schedule and reduce costs. In the area of wider applicability SMR designs and sizes are better suited for partial or dedicated use in non-electrical applications such as providing heat for industrial processes, hydrogen production or sea-water desalination. Process heat or cogeneration results in significantly improved thermal efficiencies leading to a better return on investment. Some SMR designs may also serve niche markets, for example to burn nuclear waste.

Booklets on the status of SMR technology developments have been regularly published. The objective is to provide Member States with a concise overview of the latest status of SMR designs. This booklet is reporting the advances in design and technology developments of SMRs of all the major technology lines within the category of SMRs. It covers land based and marine based water-cooled reactors, high temperature gas cooled reactors, liquid metal, sodium and gas-cooled fast neutron spectrum reactors and molten salt reactors.

This booklet is intended as a supplement to the IAEA Advanced Reactor Information System (ARIS), which can be accessed at http://aris.iaea.org. Other recent booklets published in support of ARIS are listed in Annex V.

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 officers responsible for this publication were S. Banoori, M. H. Subki and F. Reitsma of the Division of Nuclear Power.

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INTRODUCTION

The IAEA's Department of Nuclear Energy within its structure contains the Section for Nuclear Power Technology Development that is tasked to facilitate efforts of Member States in identifying key enabling technologies in the development of advanced reactor lines and addressing their key challenges in near term deployment. By establishing international networks and ensuring coordination of Member State experts, publications on international recommendations and guidance focusing on specific needs of newcomer countries are issued.

There is increasing interest in small modular reactors (SMRs) and their applications. SMRs are newer generation reactors designed to generate electric power up to 300 MW, whose components and systems can be shop fabricated and then transported as modules to the sites for installation as demand arises. Most of the SMR designs adopt advanced or even inherent safety features and are deployable either as a single or multi-module plant. SMRs are under development for all principal reactor lines: water cooled reactors, high temperature gas cooled reactors, liquid-metal, sodium and gas-cooled reactors with fast neutron spectrum, and molten salt reactors. The key driving forces of SMR development are fulfilling the need for flexible power generation for a wider range of users and applications, replacing ageing fossil-fired units, enhancing safety performance, and offering better economic affordability.

Many SMRs are envisioned for niche electricity or energy markets where large reactors would not be viable. SMRs could fulfil the need of flexible power generation for a wider range of users and applications, including replacing aging fossil power plants, providing cogeneration for developing countries with small electricity grids, remote and off grid areas, and enabling hybrid nuclear/renewables energy systems. Through modularization technology, SMRs target the economics of serial production with shorter construction time. Near term deployable SMRs will have safety performance comparable or better to that of evolutionary reactor designs.

Though significant advancements have been made in various SMR technologies in recent years, some technical issues still attract considerable attention in the industry. These include for example control room staffing and human factor engineering for multi-module SMR plants, defining the source term for multi-module SMR plants with regards to determining the emergency planning zone, developing new codes and standards, and also load-following operability aspects. Some potential advantages of SMRs like the elimination of public evacuation during an accident or a single operator for multiple modules are being challenged by regulators. Furthermore, although SMRs have lower upfront capital cost per unit, their generating cost of electricity will probably be substantially higher than that for large reactors.

Currently there are at least 50 SMR designs under development for different application. Three industrial demonstration SMRs are in advanced stage of construction: in Argentina (CAREM, an integral PWR), in China (HTR-PM, a high temperature gas cooled reactor) and in the Russian Federation (KLT40s, a floating power unit). They are scheduled to start operation between 2017 and 2020. Dozens of SMR designs are also being prepared for near term deployment. Given the above mentioned issues and challenges to be resolved, realistically the first commercial fleet of SMRs is expected to start up in the time frame of 2025 – 2030. However, large fleet deployment of SMRs will only take place beyond 2030.

To assist the reader to easily understand the current status of deployment Table 1 lists all the SMR designs with the applicable technology along with the output capacity, type of reactor and design institute information.

Table 1: Status of Deployment of SMR Designs and Technologies

Design	Output	Type	Designers	Country	Status
KLT-40S	70	Floating NPP	OKBM Afrikantov	Russian Federation	Under construction
HTR-PM	210	HTGR	INET, Tsinghua University	China	Under construction
CAREM	30	PWR	CNEA	Argentina	Under construction
ACP100	100	PWR	CNNC	China	Conceptual Design
CAP150/200	150/200	PWR	CGNPC	China	Conceptual Design
AHWR-300	300	PHWR	BARC	India	Conceptual Design
IRIS	335	PWR	IRIS Consortium	Multi Countries	Conceptual Design
DMS	300	BWR	Hitachi GE	Japan	Conceptual Design
IMR	350	PWR	MHI	Japan	Conceptual Design
SMART	100	PWR	KAERI	Republic of Korea	Certified Design
UNITHERM	6.6	PWR	NIKIET	Russian Federation	Conceptual Design
KARAT-45/100	45/100	BWR	NIKIET	Russian Federation	Conceptual Design
ELENA	68 kW	PWR		Russian Federation	Conceptual Design
RUTA-70	70 MW(th)	PWR	NIKIET	United States of America	Conceptual Design
NuScale	50 × 12	PWR	NuScale Power	United States of America	Under Development
mPower	195 × 2	PWR	BWX Technologies	United States of America	Under Development
W-SMR	225	PWR	Westinghouse	United States of America	Conceptual design completed
SMR-160	160	PWR	Holtec International	United States of America	Conceptual Design
ACPR50S	60	PWR	CGNPC	China	Conceptual Design completed
Flexblue	165	Immersed NPP	DCNS	France	Conceptual Design
RITM-200	50 × 2	Floating NPP	OKBM Afrikantov	Russian Federation	Detailed Design
VBER-300	325	Floating NPP	OKBM Afrikantov	Russian Federation	Licensing Stage
ABV-6E	6	Floating NPP	OKBM Afrikantov	Russian Federation	Final design
SHELF	6.4	Immersed NPP	NIKIET	Russian Federation	Detailed Design Underway
GTHTR300	300	HTGR	JAEA	Japan	Basic Design
GT-MHR	285	HTGR	OKBM Afrikantov	Russian Federation	Preliminary Design completed
MHR-T	205.5x4	HTGR	OKBM Afrikantov	Russian Federation	Conceptual Design
MHR-100	25 – 87	HTGR	OKBM Afrikantov	Russian Federation	Conceptual Design
PBMR-400	165	HTGR	PBMR SOC Ltd	South Africa	Preliminary Design completed
HTMR-100	35	HTGR	Steenkampskraal Thorium Limited (STL)	South Africa	Advanced Conceptual Design phase

Design	Output	Type	Designers	Country	Status
SC-HTGR	272	HTGR	AREVA	United States of America	Conceptual Design
Xe-100	35	HTGR	X-energy LLC	United States of America	Conceptual Design
LEADIR-PS	39	LMFR	Northern Nuclear Industries Incorporated	Canada	Conceptual Design
4S	10	LMFR	Toshiba Corporation	Japan	Detailed Design
BREST-OD-300	300	LMFR	NIKIET	Russian Federation	Detailed Design
SVBR-100	100	LMFR	JSC AKME Engineering	Russian Federation	Detailed Design
G4M	25	LMFR	Gen4 Energy Inc.	United States of America	Conceptual Design
EM^2	265	GMFR	General Atomics	United States of America	Conceptual Design
IMSR	185-192	MSR	Terrestrial Energy	Canada	Conceptual Design
MSTW	100	MSR	Seaborg Technologies	Denmark	Conceptual Design
ThorCon	250	MSR	Martingale	International Consortium	Conceptual Design
FUJI	200	MSR	International Thorium Molten-Salt Forum: ITMSF	Japan	Conceptual Design Completed
Stable Salt Reactor	37.5×8	MSR	Moltex Energy	United Kingdom	Conceptual Design
SmAHTR	125 MW(th)	MSR	ORNL	United States of America	Pre-conceptual design
LFTR	250	MSR	Flibe Energy	United States of America	Conceptual Design
Mk1 PB-FHR	100	MSR	University of California, Berkeley	United States of America	Pre-Conceptual Design

Figure 1 shows a pictorial view of all the Member States with Small Modular Reactors designs and technologies included in the booklet.

Additional summaries are also provided. Annex I provides the estimated timeline of the possible commercial deployment for the different SMR designs. This is based on the current design and licensing status but is of course also dependent on project financing and other factors. The reactors under construction have the highest probability for earlier deployment.

Annex II provides a summary as a function of different power ranges while Annex III shows SMR designs based on the core exit coolant temperature. This may be helpful to select designs for specific power or process heat applications, further illustrated in Annex IV for different non-electric applications.

This booklet is a supplement to the IAEA Advanced Reactor Information System (ARIS, http://aris.iaea.org). Other recent booklets published in support of ARIS are listed in Annex V.

Annex VI contains a list of commonly used acronyms.

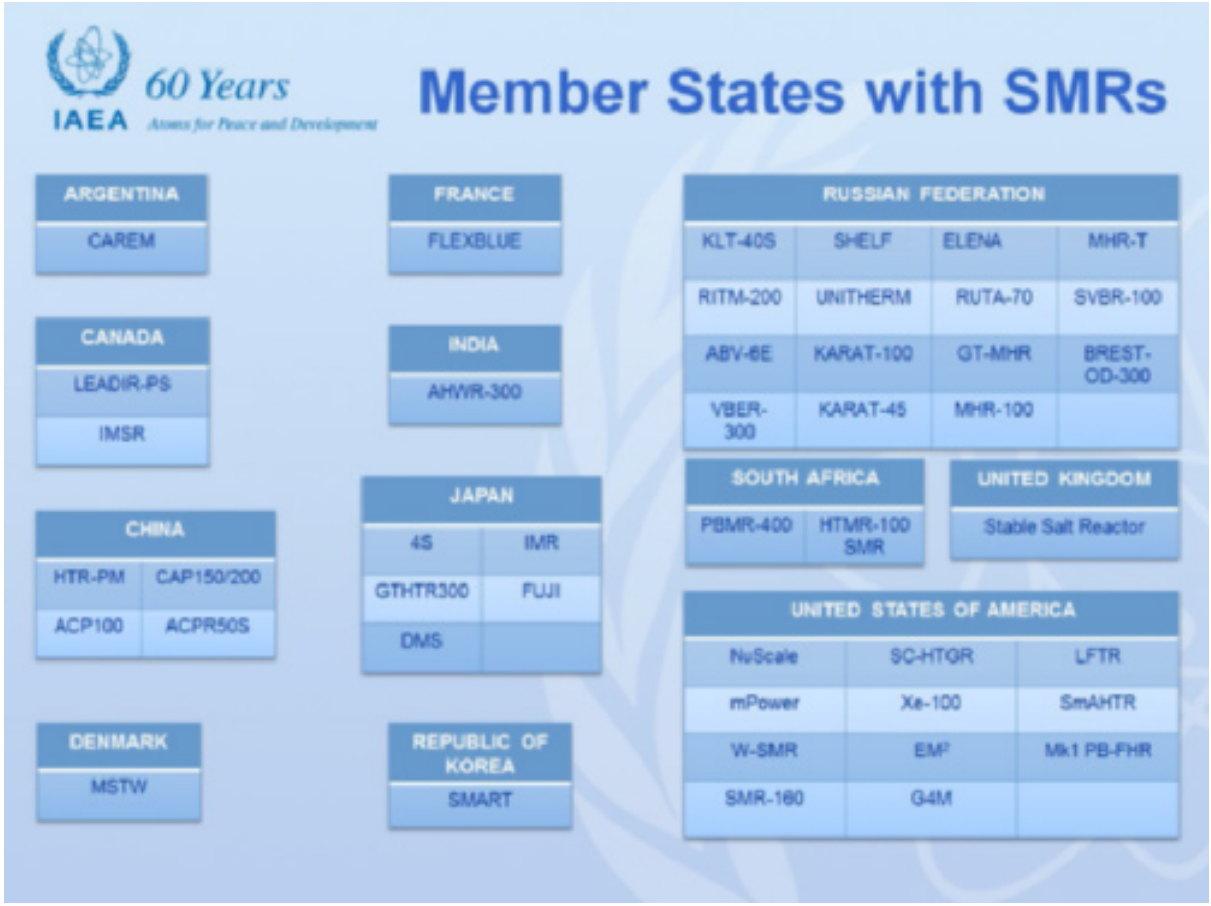


Figure 1: Member States with Small Modular Reactors

This booklet provides a brief introductory information and technical description of the key SMR designs and technologies under different stages of development and deployment. The 2016 edition of the booklet includes the SMR designs with a fast neutron spectrum and for the first time molten salt reactors (MSRs) designs have also been added. This booklet comprises five (5) parts arranged in the order of the different types of coolants and neutron spectrum adopted:

Part One (Land-based water-cooled SMRs) presents the key SMR designs adopting integral light water reactor (LWR) technologies.

The Central Argentina de Elementos Modulares (**CAREM**) reactor under development is a small, integral type pressurized light water reactor (PWR) design, with all primary components located inside the reactor vessel and an electrical output of 150–300 MW(e). Site excavation work for a 27 MW(e) CAREM-25 prototype was completed at the end of August 2012 and construction formally began in February 2014. 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 to receive certification. In September 2015, a pre-project engineering agreement was signed for the deployment of a SMART reactor in Saudi Arabia.

The China National Nuclear Corporation (CNNC) is developing the **ACP100** design and has undertaken the IAEA Generic Reactor Safety Review (GRSR) in 2015. An industrial

demonstration plant with two modules of ACP100 is planned to be deployed in Fujian Province. The Shanghai Nuclear Engineering Research and Design Institute (SNERDI), a subsidiary of the State Nuclear Power Technology Corporation (SNPTC), is developing the China Advanced Passive pressurized water reactors **CAP150** and **CAP200** The concept designs adopted passive engineered safety features and simplified systems compared to current operating PWRs, to generate electric power of 150 MW(e) and 200MW(e) respectively.

For heavy-water cooled SMRs, India has developed the advanced heavy water reactor (**AHWR300-LEU**) to generate 304 MW(e). The design incorporates vertical pressure tubes, low enriched uranium and thorium fuel, and passive safety features. The basic design has been completed. The IRIS international consortium has completed the basic design of the **IRIS** 325 MW(e) integral PWR. In Japan 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.

In the Russian Federation 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. NIKIET is also working on **KARAT-45**, a small boiling water reactor (BWR) with a rated power of 45 MW; **KARAT-100**, an integral type multi-purpose BWR with a power output of 360 MW(th) and a rated electrical output of 100 MW; and **RUTA-70** a water-cooled water-moderated integral pool-type heating reactor with a thermal capacity of 70 MW(th) dedicated for district heating or potable water production. The Kurchatov Institute is proposing the **ELENA** conceptual design, a nuclear thermoelectric plant to supply 68 kW(e) of electricity and 3.3 MW(th) of heating capacity for 25 year without refuelling.

Four integral pressurized water SMRs are under development in the USA: NuScale, mPower, the Westinghouse SMR and SMR-160. **NuScale** is a nuclear power plant made up of twelve modules each producing 50 MW(e) and design certification application is expected to be submitted to the US NRC by end of 2016, followed by application for a combined construction and operation licence (COL). Its design certification target is 2020, for the first plant that is to be built in Idaho. The **mPower** design consists of two 195 MW(e) modules. 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 preliminary design is expected to be completed by the end of 2017.

Part Two (Marine-based water-cooled SMRs) discusses concepts such as the KLT-40S (under construction), ACPR50S, and Flexblue.

In the Russian Federation, a barge mounted floating nuclear power plant (NPP) with two modules of **KLT-40S**, a compact PWR with a capacity of 35 MW(e) per module and used for cogeneration of process heat and electricity, is under construction and its commercial startup is expected in 2019. The **ACPR50S** is a small modular offshore floating reactor developed by the China General Nuclear Power Group (CGNPC). It is intended as a potential optimal solution for combined supply of heat, electricity and fresh water for marine resource development activities, energy supply and emergency support on islands and along the

coastal area. It is currently in a preliminary design stage with commissioning and connection to grid expected in 2020. In France, the **Flexblue** design is a small seabed nuclear power plant with an output of 160 MW(e). Technical discussions have been initiated with the French technical safety authority.

In the Russian Federation the **ABV-6E** is being developed by OKBM Afrikantov with an electrical output of 6 MW(e) in cogeneration mode. It 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 multipurpose nuclear icebreaker, floating or land-based NPP, is designed to generate 50 MW(e). Two serial RITM-200 icebreakers will be commissioned in 2020. The **VBER-300** is a multipurpose medium-sized power reactor with a rated electric power of 325 MW intended for land-based nuclear power plants (NPPs), nuclear cogeneration plants, and transportable floating nuclear power plants (FNPPs). The N.A. Dollezhal Research and Development Institute of Power Engineering (NIKIET) is designing the **SHELF**, a 6 MW(e) underwater, remotely operated power source.

Part Three (High Temperature Gas Cooled SMRs) provides information on the HTR-PM design under construction, other HTGR designs under development for potential certification, and details on the fundamental safety and design strategies.

In the area of innovative reactors, 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 more efficient electricity generation, a variety of industrial applications as well as for cogeneration. This enables HTGRs to 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 and the major equipment (RPV, core barrel etc.) have recently been delivered at site, with operation expected 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. One of these designs 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 the 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. Finally, the **MHR-100** prismatic modular helium reactor design of 215 MW(th) is used in multiple configurations for electricity and cogeneration. 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.

The South African developed Pebble Bed Modular Reactor (**PBMR-400**) design can produce electricity of 165 MW(e) at high efficiency via a 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 project was stopped in 2010 but the design information is secured and maintained. Also in South Africa the Steenkampskraal Thorium Limited (STL) company is aiming to finish the conceptual design of the **HTMR-100** pebble bed design by 2018 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 the 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.

Part Four (Fast Neutron Spectrum SMRs) presents the SMRs with fast neutron spectrum with all the different coolant options.

In the booklet on "Status of Innovative Fast Reactor Designs and Concepts" (see Annex V) the four major fast reactor options were described. They are sodium cooled fast reactor (SFR), the heavy liquid metal-cooled (HLMC) fast reactor, the gas-cooled fast reactor (GFR) and molten salt fast reactor (MSFR). In this booklet only fast reactor designs with the first three types of coolant are included. The MSR designs, all with thermal neutron spectrum, are included in part five.

The Lead-cooled Integral Reactor-Passively Safe (**LEADIR-PS**), a small nuclear reactor being developed by the Northern Nuclear Industries Incorporated (N^2I^2) is intended to provide diverse electricity and process heat demands of Canada. LEADIR-PS plants under development are LEADIR-PS30 (30 MW(th), up to 11 MW(e)), LEADIR-PS100 (100 MW(th), up to 39 MW(e), and LEADIR-PS300 (300 MW(th), up to 120 MW(e)). The **4S** (super-safe, small and simple) is a small sodium cooled fast reactor without on-site refuelling. Being developed by Toshiba Corporation, Japan as distributed energy source for multi-purpose applications, the 4S offers two power outputs of 30 MW(th) and 135 MW(th), respectively.

In the Russian Federation the N.A. Dollezhal Research and Development Institute of Power Engineering (NIKIET) is designing the **BREST-OD-300** lead cooled fast reactor that uses a two circuit heat transport system to deliver heat to a supercritical steam turbine and generate electricity of 300 MW. The BREST-OD-300 power unit is designed as a pilot and demonstration unit and is considered the prototype of a fleet of medium sized lead cooled fast reactors. The **SVBR-100** is a multipurpose small modular fast reactor lead–bismuth (LBE) cooled being developed by JSC AKME Engineering, Russian Federation, with an equivalent

electric power of 100 MW. Physical and power generation start-up of its pilot plant is expected in 2019.

From the United States of America two fast reactor designs are included. The **G4M** conceptual design is being developed by the Gen4 Module (G4M) Inc. It is a low capacity, portable and self-contained reactor designed to generate 25 MW(e). It uses lead–bismuth eutectic as the primary coolant. The G4M does not need a human operator and employs a fuel cycle of 10 years with no on-site refuelling. Niche markets include producing steam/ electricity for mining/oil and gas production sites, isolated and island communities and government facilities. The Energy Multiplier Module (**EM^2**) is a helium-cooled fast reactor developed by General Atomics. It is designed as a modular, grid-capable power source with a net unit output of 265 MW(e). With a core outlet temperature of 850°C, it can also be used as a high temperature process heat source. The reactor employs a "convert and burn" core design which converts fertile isotopes to fissile and burns them in situ over a 30-year core life.

Part Five (Molten Salt SMRs) presents the SMRs that utilize molten salt fuelled (and cooled) advanced reactor technology. Molten salt reactors (MSRs) have several benefits and include (somewhat dependent on the specific design, and relative to currently deployed water cooled reactors: (i) higher operating temperatures leading to increased overall efficiencies; (ii) low coolant pressure; (iii) reduced volume and lifetime of high level waste; (iv) salient safety characteristics; (v) in the case of MSRs with molten fuel, elimination of technical challenges associated with solid fuel related to high burn-up effects; and (vi) flexibility of fuel cycles (uranium, plutonium, thorium).

The Integral Molten Salt Reactor – 400 Megawatts-thermal (**IMSR400**) is a small modular molten salt fuelled reactor developed by Terrestrial Energy Canada. It electrical capacity is ~190 MW(e). All key components of IMSR are permanently sealed during operation and replaced completely, as a single unit, at the end of its useful service life (nominally 7 years). Pre-licensing vendor design review of the IMSR is currently underway. The Seaborg Technologies' Molten Salt Thermal Wasteburner (**MSTW**) is a concept with a single molten fuel-salt to be operated on a combination of spent nuclear fuel and thorium. The design efforts are currently focused on advanced multi-physics, neutronics, and early engineering details.

ThorCon is a thermal spectrum molten salt nuclear reactor that utilizes 250 MW(e) power modules. Each module contains two replaceable reactors in sealed Cans . When one reactor is producing power the other is in cooldown mode. Cans are replaced every four years when the fuelsalt is transferred to the new Can. The low pressure and short 4-year Can plant life means ThorCon plans to use normal piping thicknesses and easily automated, ship-style steel plate construction methods to reduce costs. Currently a pre-feasibility study is being done in Indonesia. Under development in Japan, through the International Thorium Molten-Salt Forum, the **FUJI** concept design can be used for transmutation of Pu and/or minor actinides. It aims not only to generate electricity of 200 MW(e) but also heat for hydrogen production (it has a high exit temperature of 704 °C) and water supply by desalination of seawater.

The Stable Salt Reactor (**SSR**), being developed by Moltex Energy in the United Kingdom is unique in its use of molten salt fuel replacing solid pellets in conventional solid fuel assemblies. This brings the major advantages of safe molten salts without the technical hurdles of managing a mobile liquid fuel. The proposed power plant will deliver 300MWe

made up by eight 37.5MWe modules. The conceptual design has been completed and the key claims have been independently validated.

The MSR concept was originally developed in the USA headed by the Oak Ridge National Laboratory (ORNL). The ORNL program culminated in the construction and successful operation of the Molten Salt Reactor Experiment (MSRE). The MSRE employed a molten fluoride fuel salt, which was also the primary coolant. Three MSR designs from the USA are included.

SmAHTR is a small 125 MW(th) fluoride salt–cooled high-temperature reactor (FHR) design concept being developed by ORNL intended to match the energy requirements of coupled industrial processes (for efficient hydrogen production as example). It is not intended for electricity generation. The purpose is to enable an integrated assessment of the potential performance of deliberately small FHR designs and to provide guidance to the overall FHR research and development effort.

The liquid-fluoride thorium reactor (**LFTR**) design by Flibe Energy is a graphite-moderated, thermal-spectrum reactor with solutions of liquid fluoride salts containing both fissile and fertile materials. The fuel and blanket salts are kept separated in two plena integrated into a single structure within the reactor vessel. A closed-cycle gas turbine power conversion system is proposed for 250 MW(e) electrical generation with the objective to produce electricity at low cost by efficiently consuming thorium. The Mark 1 Pebble-Bed Fluoride-Salt-Cooled High-Temperature-Reactor (**Mk1 PB-FHR**) is a small, modular graphite-moderated reactor. It is being developed by University of California, Berkeley. The design is differentiated from other MSRs because it use coated particle fuels, cooled by the fluoride salt flibe. The power plant is designed to produce 100 MW(e) of base-load electricity when operated with only nuclear heat, and to increase this power output to 242 MW(e) using gas co-firing for peak electricity generation.

In this booklet, effort has been made to present all SMR designs within the above five parts. Each description includes a general design description and philosophy, target applications, development milestone, nuclear steam supply system, a comprehensive table of the major design parameters, and then descriptions of the reactor core, engineered safety features, plant arrangement, design and licensing status, and finally plant economics. Not all small reactor designs presented can strictly be categorized as small modular reactors. Some strongly rely on proven technologies of operating large capacity reactors, while others do not use a modular or integral design approach. They are presented in this booklet for reason of completeness and since designers foresee certain niche markets for their products.

The 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.

It is hoped that this Booklet will be useful to Member States with a general interest in SMRs, as well as to those newcomer countries looking for more specific technical information. It should also further promote contributions to and the use of the IAEA Advanced Reactor Information System (ARIS).

WATER COOLED SMALL MODULAR REACTORS (LAND BASED)

CAREM (CNEA, Argentina)

1. Introduction

CAREM is a national SMR development project based on LWR technology coordinated by the Argentina's 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 level of safety. CAREM 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. CAREM reactor was developed using domestic technology, at least 70% of the components and related services for CAREM were sourced from Argentinean companies. The prototype module of CAREM (CAREM-25) has a thermal power of 100 MW (30 MW(e)) is currently under construction.

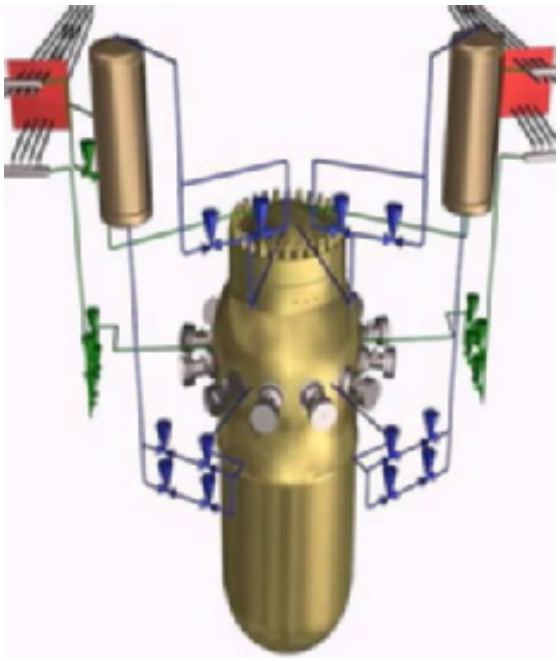


Figure 1: CAREM module with its second shutdown and residual heat removal systems (Reproduced courtesy of CNEA)

2. Target Applications

CAREM 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.

3. Development Milestones

1984	CAREM concept was presented in Lima, Peru, during the IAEA Conference on SMRs and was one of the first of the 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 and contracts and agreements between stakeholders are under discussion
2012	Civil engineering works
2014	8 February, formal start of construction
2019	First fuel load

4. General Design Description

Design Philosophy

CAREM is a natural circulation based indirect-cycle reactor with distinctive features that simplify the design and improving safety performance. Its primary circuit is fully contained in the reactor vessel and it does not need any primary recirculation pumps. The pressurization is achieved by balancing vapour production and condensation in the vessel. The CAREM design reduces the number of sensitive components and potentially risky interactions with the environment.

Some of the significant design characteristics are:

- integrated primary cooling system;
- self-pressurized;
- core cooling by natural circulation;
- in-vessel control rod drive mechanisms;
- safety systems relying on passive features;
- Balanced and optimized design with a cost-effective internalization of safety.

Nuclear Steam Supply System

CAREM is an integrated reactor. Its high-energy primary system (core, steam generators, primary coolant and steam dome) is contained inside a single pressure vessel. Primary cooling flow is achieved by natural circulation, which is induced by placing the steam generators above the core as shown in *Figure 1*. Water enters the core from the lower plenum. After being heated, the coolant exits the core and flows up through the chimney to the upper steam dome. In the upper part, water leaves the chimney through lateral windows to the external region. It then flows down through modular steam generators, decreasing its enthalpy. The coolant exits the steam generators and flows through the downcomer to the lower plenum, closing the circuit. At steady-state conditions, the driving forces created by the density differences along the circuit are balanced by the friction and form losses, so there is an adequate flow rate in the core. The coolant also acts as a neutron moderator.

Reactor Core

The reactor core of CAREM-25 has fuel assemblies of hexagonal cross section. There are 61 fuel assemblies having about 1.4 meters active length. Each fuel assembly contains 108 fuel rods with 9 mm outer diameter, 18 guide thimbles and one instrumentation thimble as shown in *Figure. 2*. The fuel is 1.8% - 3.1% enriched UO_2.The fuel cycle can be tailored to customer requirements, with a reference design for the prototype of 510 full-power days and 50% of core replacement.

Reactivity Control

Core reactivity is controlled by the use of Gd_2O_3 as burnable poison in specific fuel rods and movable absorbing elements belonging to the adjustment and control system. Neutron poison in the coolant is not used for reactivity control during normal operation and in reactor shutdown. Each absorbing element consists of a cluster of rods linked to a structural element

('spider'), so the whole cluster moves as a single unit. Absorber rods fit into the guide tubes. The absorbent material is the commonly used Ag-In-Cd alloy. Absorbing elements are used for reactivity control during normal operation, and to produce a sudden interruption of the nuclear chain reaction when required.

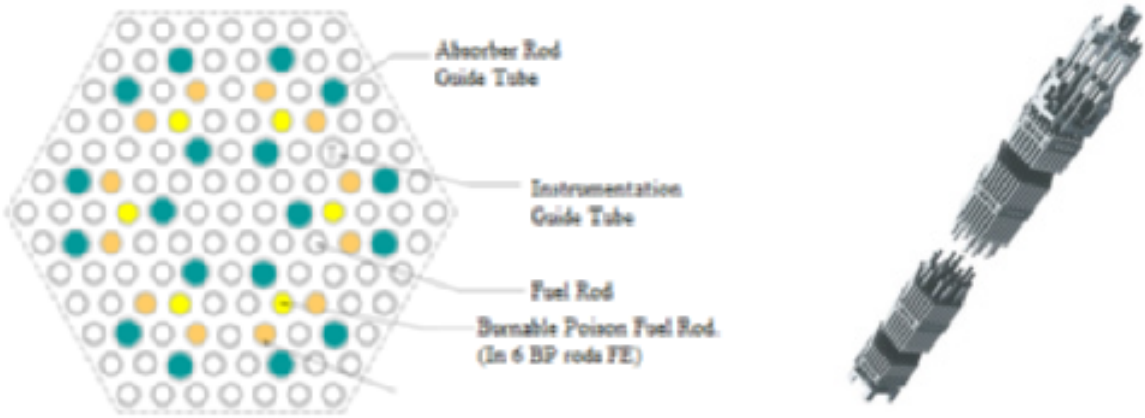


Figure 2: Reactor Core and of Fuel Assembly of CAREM-25
(Reproduced courtesy of CNEA)

Reactor Pressure Vessel and Internals

Reactor Pressure Vessel (RPV) of CAREM-25 has height of 11 meters and 3.4 meters diameter having variable thickness of 13 cm to 20 cm. The RPV is built in forged steel with internal lining of stainless steel.

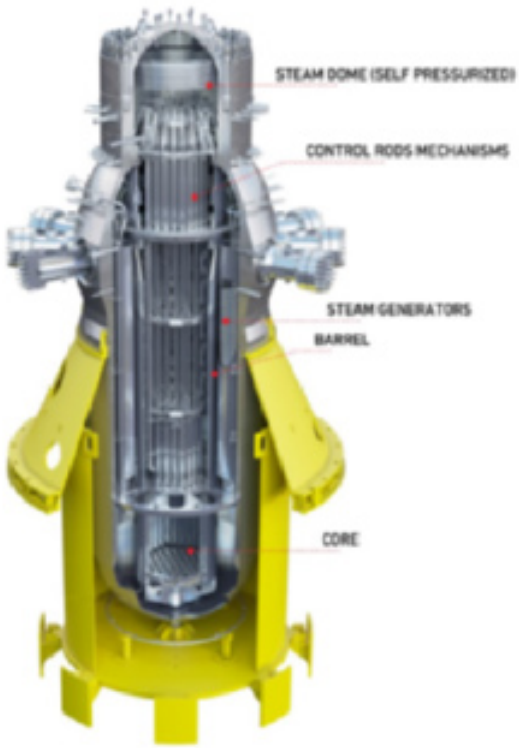


Figure 3: CAREM-25 Reactor Vessel and Internal
(Reproduced courtesy of CNEA)

Reactor Coolant System

Primary cooling in CAREM is achieved using natural circulation. The primary circuit of CAREM is fully contained in the reactor vessel and it has no cooling pumps. The reactor core heats reactor coolant causing the coolant to flow upward through the riser. When the heated reactor coolant exits the riser, it passes over the tubes of the helical coil steam generators, which act as a heat sink.

Pressurizer

Self-pressurization of the primary system in the steam dome is the result of the liquid–steam equilibrium. The large steam volume in the RPV acting as an integral pressurizer, also contributes to damping of eventual pressure perturbations. Due to self-pressurization, bulk temperature at core outlet corresponds to saturation temperature at primary pressure. In this way, typical heaters present in conventional PWRs are eliminated.

Steam Generator

In CAREM-25, twelve identical mini-helical vertical steam generators, of the once-through type are placed equidistant from each other along the inner surface of the RPV. Each of them consists of a system of 7 coiled piping layers, 52 parallel pipes of 26m length. They are used to transfer heat from the primary to the secondary circuit, producing superheated dry steam at 4.7 MPa. The secondary system circulates upwards within the tubes, while the primary coolant moves in counter-current flow. In order to achieve a mostly uniform pressure-loss and superheating on the secondary side, the length of all tubes is equalized. Due to safety reasons, the steam generators are designed to withstand the primary pressure without pressure in the secondary side and the whole live steam system is designed to withstand primary pressure up to isolation valves (including the steam outlet/water inlet headers) in case of SG tube breakage.

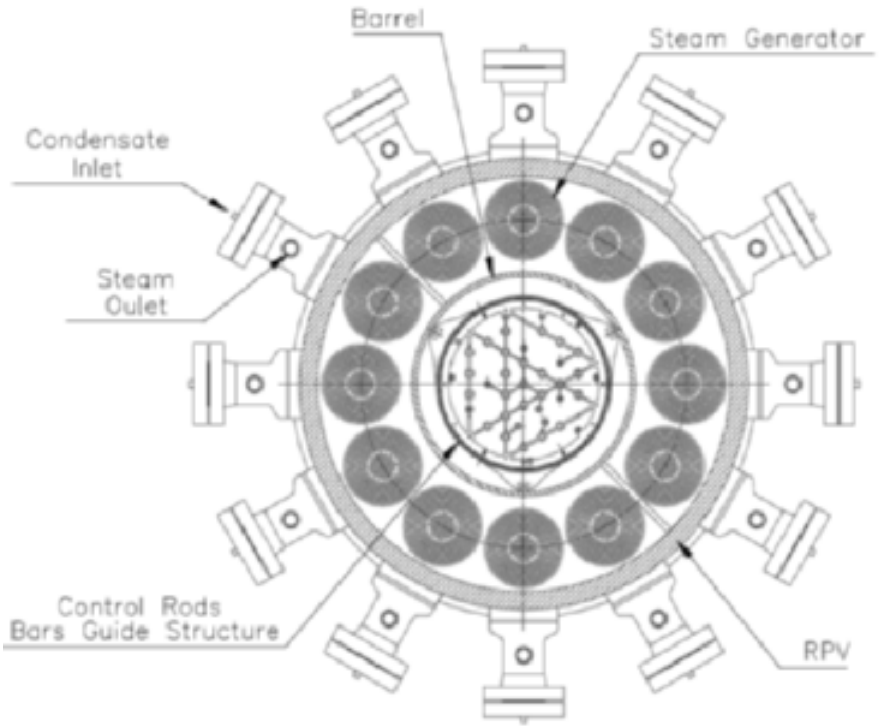


Figure 4: Layout of CAREM-25 steam generators inside RPV (Reproduced courtesy of CNEA)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology Developer	CNEA
Country of origin	Argentina
Reactor type	Integral PWR
Electrical capacity (MW(e))	~30
Thermal capacity (MW(th))	100
Expected Capacity Factor (%)	> 90
Design Life (years)	60
Plant Footprint (m^2)	Not available
Coolant/moderator	Light water
Primary circulation	Natural circulation
System pressure (MPa)	12.25
Core inlet/exit temperatures (^oC)	284 / 326
Main Reactivity Control Mechanism	Control Rod Driving Mechanism (CRDM) only
RPV Height (m)	11
RPV Diameter (m)	3.2
RPV, internals and SGs Weight (metric ton)	267
Configuration of Reactor Coolant System	Integrated
Power Conversion Process	Indirect Rankine Cycle
Fuel Type/Assembly Array	UO_2 pellet/hexagonal
Fuel Assembly 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)
Cogeneration Capability	Possible
Approach to Engineered Safety Systems	Passive
Number of safety trains	2
Refuelling Outage (days)	not available
Distinguishing features	Core heat removal by natural circulation, Pressure suppression containment
Modules per Plant	1
Target Construction Duration (months)	~36
Seismic design (g)	0.25
Predicted core damage frequency (per reactor year)	~1E7
Design Status	Under construction (as prototype)

5. Safety Features

Engineered Safety System Configuration and Approach

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

independence, separation and diversification and act automatically. Each can maintain the core sub-critical in all shut down states; First Shutdown System (FSS) consists of 9 fast shutdown rods and 16 reactivity adjust and control rods located over the core. They fall by gravity in case of necessity. Second Shutdown System consists of a gravity driven injection device of borated water (two tanks) at high pressure which actuates automatically when failure of the FSS is detected. For the grace period of 36 hours, core decay heat removal can ensure safe core temperature due to availability of one out of two PRHRS in the case of loss of heat sink or Station Black-out (SBO). In CAREM, SBO is classified as a design basis event. 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. Through natural circulation, the design provides core decay heat removal, transferring it to dedicated pools inside the containment and then to the suppression pool. Two redundant diesels provide emergency supply for active cooling systems for the long term. Despite the low frequency of a SBO longer than 36 hours, prevention is considered for grace period prolongation by 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. Safety classification of systems, structures and components (SSCs) important to safety is based on identification of low level safety functions (LLSF) -derived from the Fundamental Safety Functions- and safety functional groups of SSCs that fulfil those functions. Criteria for safety categories assignation to LLSF and classes to SSC are obtained from the way the principle of Defence in Depth is internalized in the design, and probabilistic and deterministic considerations. Three categories and classes are defined. This methodology, in accordance with IAEA SSG-30, provides a clear assignation of design rules and requirements to systems important to safety and its SSCs.

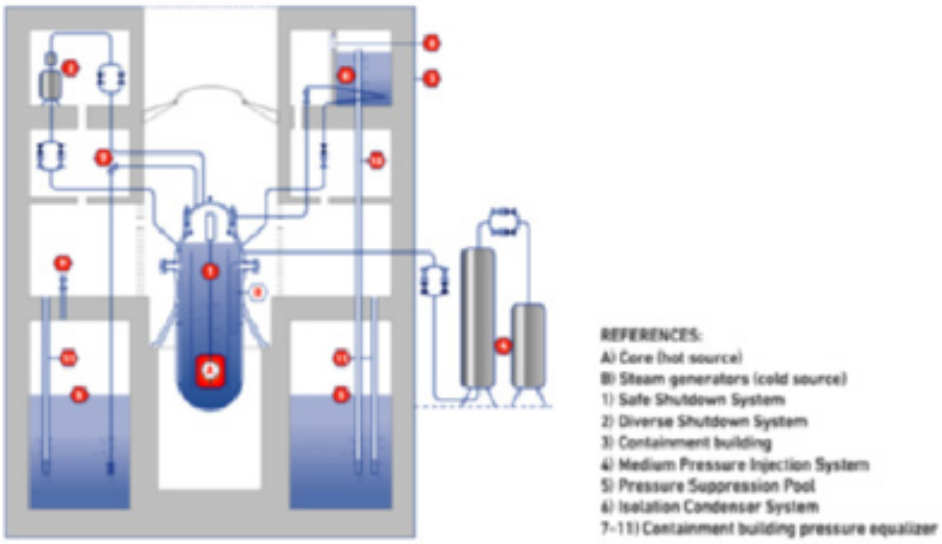


Figure 5: Safety Systems of CAREM-25
(Reproduced courtesy of CNEA)

Containment System
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 0.25g. It is designed to sustain the pressure of 0.5 MPa. Ultimate heat sink

inside the containment during the grace period provides protection for external extreme events.

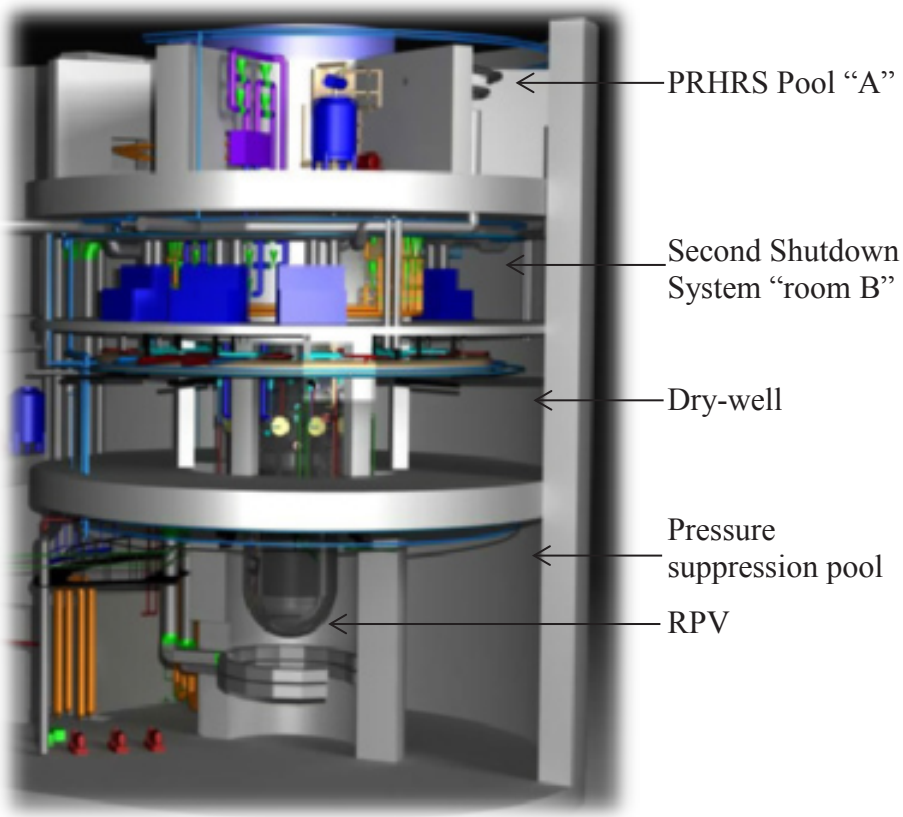


Figure 6: CAREM-25 containment cut-away view (Reproduced courtesy of CNEA)

6. Plant Safety and Operational Performances

The natural circulation of coolant produces different flow rates in the primary system according to the power generated or removed. Under different power transients, a self-correcting response in the flow rate is obtained. Due to the self-pressurizing of the RPV (steam dome), the system keeps the pressure very close to the saturation pressure. Under all operating conditions this could prove to be sufficient to guarantee a remarkable stability of the RPV pressure response. The control system is capable of keeping the reactor pressure practically at the operating set point through different transients, even in case of power ramps. The negative reactivity feedback coefficients and the large water inventory in the primary circuit combined with the self-pressurization features make this behaviour possible with minimum control rod motion. In addition, a condensation coil is included in the steam dome in order to tune the reactor operational point. Many events that would lead to accidental conditions are rendered impossible by the innovative design, like large LOCA, LOFA and control rod ejection between others. On the other hand some postulated multiple failure events are considered in the design basis as ATWS, PRHRS and ancillary diesels failure. Moreover large primary water/power ratio makes transients manageable after most severe system or component failure (large thermal inertia). Deterministic and probabilistic safety analyses were developed; they fulfil regulatory requirements and large margins against acceptance criteria are observed.

7. Instrumentation and Controls Systems

Plant control is performed by a distributed control system, computer based and with high availability. There are two diverse protection systems: First Reactor Protection System (FRPS) and Second Reactor Protection System (SRPS), each system carries four redundancies. There are two (2) diverse nuclear instrumentation systems (NIS), one each for the FRPS and SRPS.

8. Plant Arrangement

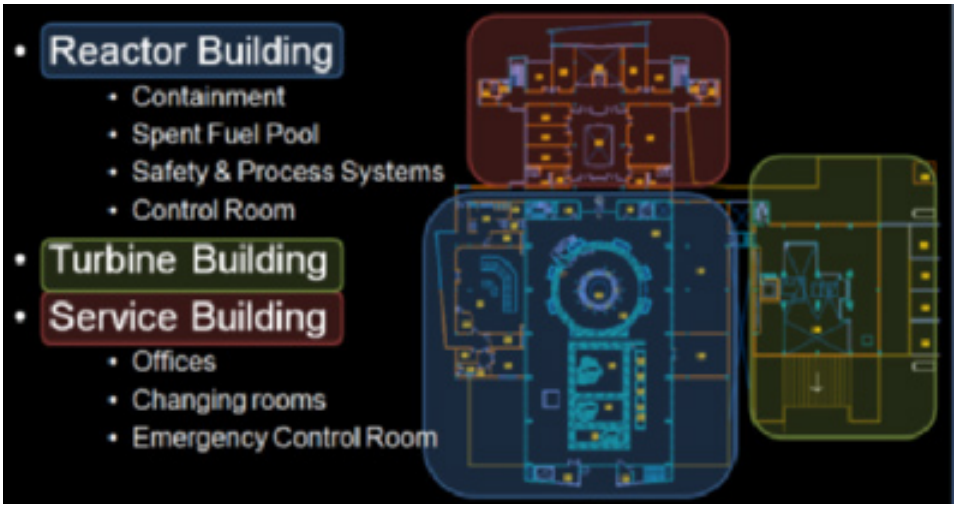


Figure 7: CAREM-25 plant layout (Reproduced courtesy of CNEA)

9. Design and Licensing Status

After completion of licensing process, construction of the Prototype started. Contracts with different Argentinean stakeholders for manufacturing of components issues have already been signed. Environmental Impact Study was approved by the Local Authority. Non-nuclear buildings first concrete was poured in February 2014.

Figure 8: CAREM-25 progressing in construction as of March 2016 (Reproduced courtesy of CNEA)

10. Plant Economics

As CAREM-25 is a prototype, plant economics is not provided.

ACP100 (CNNC, China)

1. Introduction

The ACP100 is an integrated PWR design developed by China National Nuclear Corporation (CNNC) to generate an electric power of 100 MW(e). The ACP100 is based on existing PWR technology adapting passive safety system, which in case of operational transients and postulated design basis accidents uses natural convection to cool down the reactor. The ACP100 integrated design of its reactor coolant system (RCS) enables the installation of its major primary circuit's components within the reactor pressure vessel (RPV), as shown in *Figure 1*. The ACP100 plant design allows a deployment of one to eight module(s) to generate larger plant output as demands arise. The plant design contains the following technical features:

- An integrated primary system and equipment layout, eliminating the possibility of a large break loss of coolant accidents (LOCA);
- The reactor and spent fuel pool are installed underground, as shown in *Figure 2*, providing better protection against external events, and reduced possibility of radioactive release;
- Large primary coolant inventory resulting in large thermal inertia, thus providing longer response time in the case of transients or accidents;
- Passive severe accident prevention and mitigation systems such as containment hydrogen eliminator, cavity flooding, etc. ensure the integrity of containment.

Figure 1: Reactor System Configuration of ACP100
(Reproduced courtesy of CNNC)

2. Target Application

The ACP100 is a multipurpose power reactor designed for electricity production, heating, steam production or seawater desalination and is suitable for remote areas that have limited energy options or industrial infrastructure.

3. Development Milestones

2011	CNNC signed an agreement with the Zhangzhou municipal government in Fujian Province to host the first two ACP100 demonstration units
July 2012	CNNC signed an agreement with the Chenzhou municipal government in Hunan Province to deploy an ACP100 power plant
2013	Basic design completed
December 2014	Preliminary safety assessment report (PSAR) approved
April 2015	An agreement to conduct a generic safety review for ACP100 was signed between the IAEA and CNNC

4. General Design Description

Design Philosophy

The ACP100 realizes design simplification by integrating the primary cooling system and enhanced safety by means of passive safety systems.

Nuclear Steam Supply System

The integrated nuclear steam supply system (NSSS) design of the ACP100 consists of the reactor core, sixteen (16) once-through steam generator (OTSG) and a pressurizer that are installed within the a reactor pressure vessel (RPV). The four (4) canned motor pumps are installed nozzle to nozzle to the RPV.

Reactor Core

The 57 fuel assemblies (FAs) of ACP100 core with total length of 2.15m core have a squared 17x17 configuration. The expected average fuel enrichment is about 2.4 - 4.0%. The reactor will be able to operate 24 months per fuel cycle. In August 2012, Jianzhong nuclear fuel fabrication plant (subsidiary of CNNC) at Yibin started manufacturing FAs and control rod samples for ACP100 R&D project at NPIC.

Reactivity Control

The reactivity is controlled by means of control rods, solid burnable poison and soluble boron dispersed in the primary coolant. There are 25 control rods, with a magnetic force type control rod driving mechanism (CRDM).

Reactor Pressure Vessel and Internals

The RPV and equipment layout are designed to enable the natural circulation between reactor core and steam generator. The RPV is protected by a safety relief valves against over-pressurization in the case of strong differences between core power and the power removed from the RPV.

Reactor Coolant System

The ACP100 primary cooling mechanism under normal operating condition and shutdown condition is done by forced circulation. The RCS has been designed to ensure adequate cooling of reactor core under all operational states, and during and following all postulated off normal conditions. The four (4) reactor coolant pumps (RCP) are mounted into the RPV through short stub pipes, enabling the elimination of large bore piping. The integral design of RCS significantly reduces the flow area of postulated small break LOCA.

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)	200000
Coolant/moderator	Light water
Primary circulation	Forced circulation
System pressure (MPa)	15.00
Core inlet/exit temperatures (^oC)	282.6/ 323.4
Main Reactivity Control Mechanism	Control rod drive mechanism (CRDM), solid burnable poison and boron solution
RPV Height (m)	10
RPV Diameter (m)	3.19
Module Weight (metric ton)	250
Configuration of Reactor Coolant System	Integrated
Power Conversion Process	Indirect Rankine cycle
Fuel Type/Assembly Array	UO_2/17x17 square pitch arrangement
Fuel Assembly Active Length (m)	2.15
Number of Fuel Assemblies	57
Fuel Enrichment (%)	2.40 – 4.00
Fuel Burnup (GWd/ton)	< 45000
Fuel Cycle (months)	24
Cogeneration Capability	Possible
Approach to Engineered Safety Systems	Passive
Number of safety trains	2
Refueling Outage (days)	40
Distinguishing features	Integrated reactor with tube-in-tube once through steam generator, nuclear island under the ground level
Modules per Plant	1-8
Target Construction Duration (months)	36
Seismic design (g)	0.30
Predicted core damage frequency (per reactor year)	< 1E–6
Design Status	Basic design

Steam Generator

There are sixteen (16) OTSGs, which are integrated within the RPV. All of the 16 OTSGs are fitted in the annulus between the reactor vessel and hold-down barrel. The bottoms of OTSGs are limited their position by the hole on barrel supporting hub, the heads are welded to the reactor vessel vapour cavity.

Pressurizer
The pressurizer of ACP100 is located outside of the reactor vessel. The pressurizer is a vertical, cylindrical vessel with hemispherical top and bottom heads. Electrical heaters are installed through the bottom head of the vessel, while the spray nozzle, the spring-operated safety valves and the automatic depressurization systems (1 and 2 stages) valves connection nozzles are located in the top head of the vessel.

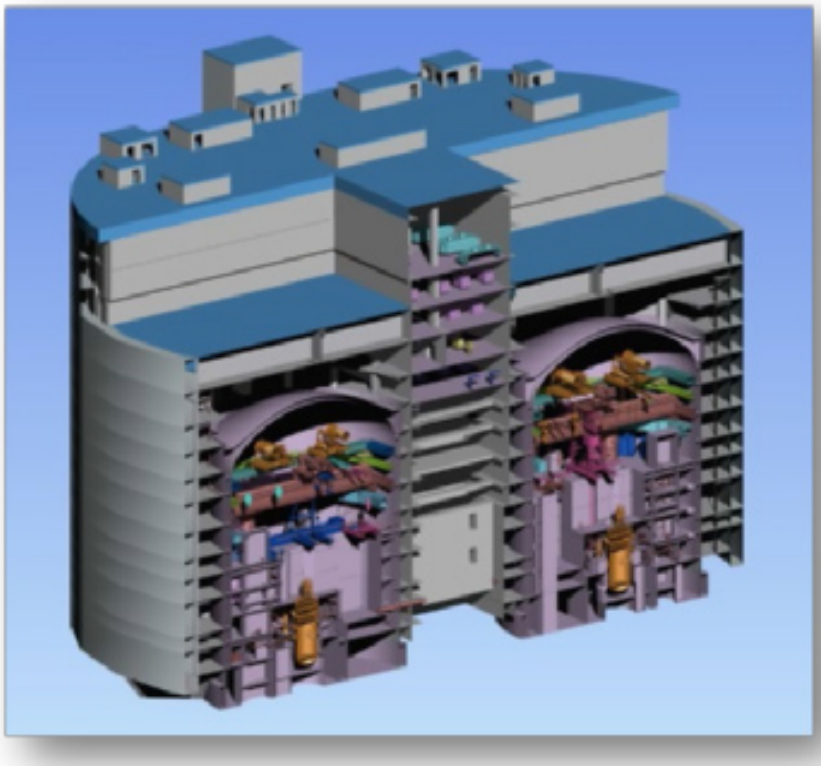


Figure 2: ACP100 Reactor Building underground location
(Reproduced courtesy of CNNC)

5. Safety Features

The ACP100 is designed with inherent and passive safety features, eliminating large bore primary coolant piping which in turn eliminates large break LOCA. The passive safety system mainly consists of the passive decay heat removal system (PDHRS), passive safety injection system, passive containment heat removal system and reactor depressurization system (RDP).

Engineered Safety System Approach and Configuration
The ACP100 is designed with several passive safety features and severe accident mitigation features. Enhanced safety and physical security of ACP100 are made possible by installing the reactor building and spent fuel pool below ground level. When the spent fuel pool is filled with spent fuel of ten (10) years, the cooling system is able to cope for seven (7) days of cooling in the case of accident before boiling dry and uncovering fuel. Severe accident prevention and mitigation are achieved through: passive reactor cavity flooding prevents RPV melt, passive hydrogen recombination system prevents containment hydrogen explosion and maintain the containment integrity under severe accidents, automatic pressure relief system and RPV off-gas system to remove non-condensable gas gathered at RPV head after accidents.

Decay heat removal system
The PDHRS prevents core meltdown in the case of design basis accident (DBA) and beyond DBA, such as station black out, complete loss of feedwater, small-break LOCA (i.e., to prevent the change of beyond DBA to severe phase). The PDHRS of the ACP100 consists of one emergency cooler and associated valves, piping, and instrumentation. The emergency cooler is located in the containment cooling water storage tank, which provides the heat sink for the emergency cooler. The PDHRS heat exchanger is installed in the in-containment water storage tank filled with coolant. The decay heat is removed from the core by natural circulation or forced circulation by the RCPs. The PDHRS provides cooling for 3 days without operator intervention or 14 days with water supply from the cooling pool by drained by gravity force.

Emergency core cooling system
The emergency core cooling system (ECCS) consists of two coolant storage tanks, two injection tanks, an in-containment water storage tank and associated injection lines. The ACP100 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 (7) days. Under accident condition, heat is being removed by gas and steam convection between containment and ultimate heat sink by natural circulation, thus ensuring the containment integrity.

Internal Refuelling Water Storage Tank
The internal refuelling water storage tank is a passive water tank, resting on the internal structure base slab. During refueling operations, it provides water for refueling cavity, internals storage compartment and refuelling transfer canal to complete the refueling operation. Under the condition of loss-of-coolant accident and the steam pipe rupture, it provides water for emergency reactor core cooling. In the severe accident, water in it floods the internal structure under the balanced water level as a result of gravity. During the operation of automatic depressurization system (ADS), it absorbs the sprayed steam from the (ADS). During the operation of the passive residual heat removal cooler, it works as the heat sink of the passive residual heat removal system.

Reactor pool
The reactor pool is used during refuelling operations or inspection of reactor internals. The reactor pool has the following two compartments which can be separated by bulkhead: reactor cavity and internals storage pool adjacent to the reactor.

Containment system
The ACP100 containment houses the RCS, the passive safety systems and the auxiliary systems. Small concrete containment with steel plate membrane encased and passive containment heat removal system are adopted. The passive containment heat removal system consists of two containment condensers, water storage tank and associated valves, piping and instrumentation.

6. Plant safety and Operational Performances

Nuclear safety is always the first priority considered in design of ACP100 SMR that can be safely operated and provide clean energy. The ultimate goal of nuclear safety is to establish and maintain an effective defence that can effectively protect people, community, and environment from radioactive disaster. To be specific, the design and operation of ACP100 ensures that radiation dose to the workers and to the members of the public do not exceed the dose limits and kept as low as reasonably achievable. Accident prevention measures ensure

that radioactive consequences are lower than limited dose in terms of all the considered accident sequence and even in the unlikely severe accident, mitigation of accident induced influences can be ensured by implementing emergency plan. The design of ACP100 incorporates operational experience of state of art design. Proven technology and equipment are adopted as much as reasonably possible.

7. Instrumentation and Control systems

The Instrumentation and Control (I&C) system design for ACP100 will be based on defence in depth concept, compliance with the single failure criterion and diversity. The diversity in the design of I&C system is achieved through: (1) different hardware and software platforms for 1E and N1E I&C, (2) the reactor protection system (RPS) with functional diversity, and (3) diverse protection systems to cope with the common mode failure of the RPS. I&C systems of the NSSS include reactor nuclear instrumentation system, RPS, diverse actuation system, reactor control system, rod control and rod position monitoring system, reactor in–core instrumentation system, loose parts and vibration monitoring system and other process control systems. The general control scheme of the NSSS is automatic control combined with manual control. The control systems have an automatic control range from 0–100% full power, except for the reactor power control system and the SG feedwater control system. The instrumentation system uses information from three (3) separate types (each type includes 4 independent channel) of instrument channel having source range, intermediate range and power range to provide three separate protection and monitoring levels. The RPS has a configuration of four (4) redundant divisions. The functions of reactor trip and ESF actuation are completed by the four (4) redundant protection divisions. Four (4) redundant measurements, using four (4) separate sensors, are made for each variable used for reactor trip.

8. Plant Arrangement

The two reactor modules share the fuel assembly building, the electrical building and nuclear auxiliary building, as shown in *Figure 3*. Thus maximizing the sharing of facilities leads to a reduced cost.

Reactor building

Reactor building and spent fuel pool of the ACP100 reactor are located under the ground level to enhance better protection from external events, reduce the radioactive material release and provide better security.

Electrical building

The electrical building contains storage battery, power distribution equipment, I&C equipment; main control room, remote shutdown station; ventilation system and smoke exhaust system in the related rooms; chilled water system is employed for the ventilation of the electrical equipment room.

Connection building

The connection building is meant to connect reactor building and other buildings, for convenience of personnel passage, equipment transportation and containment dome tensioning. There are pumps for residual heat removal system, pipes and components for main feedwater system and main steam generator system, as well as process system penetration, I&C penetrations in the connection building.

Balance of plant:

Turbine Generator building

The turbine generator building is arranged longitudinal to the main nuclear building. The head of steam turbine faces towards the nuclear building. The moisture separator re-heater (MSR) is arranged on the other side of operation layer of high pressure cylinder. The plant is mainly equipped with turbine, generator, excitation device, MSR, condenser, condensate pump, low-pressure heater, deaerator, feed pump and other auxiliary equipment.

Electric power systems

The turbine-generator (T/G) system is a double flow, tandem compound, with the last rotating blades have the length of 800 mm. The most common speed of 3000 rpm is adopted with an integrated reheat unit. The T/G train consists of a single flow combined high pressure cylinder, one double-flow low-pressure cylinders, one generator, one external MSRs with one stages of reheating are located on the left side of the high pressure cylinder. Other related system components include a complete T/G bearing lubrication oil system, a digital electro-hydraulic (DEH) control system with supervisory instrumentation, a turbine gland seal system, over speed protective devices, turning gear, a turbine-generator stator coil, a turbine-generator seal oil system, a rectifier section, and a voltage regulator.

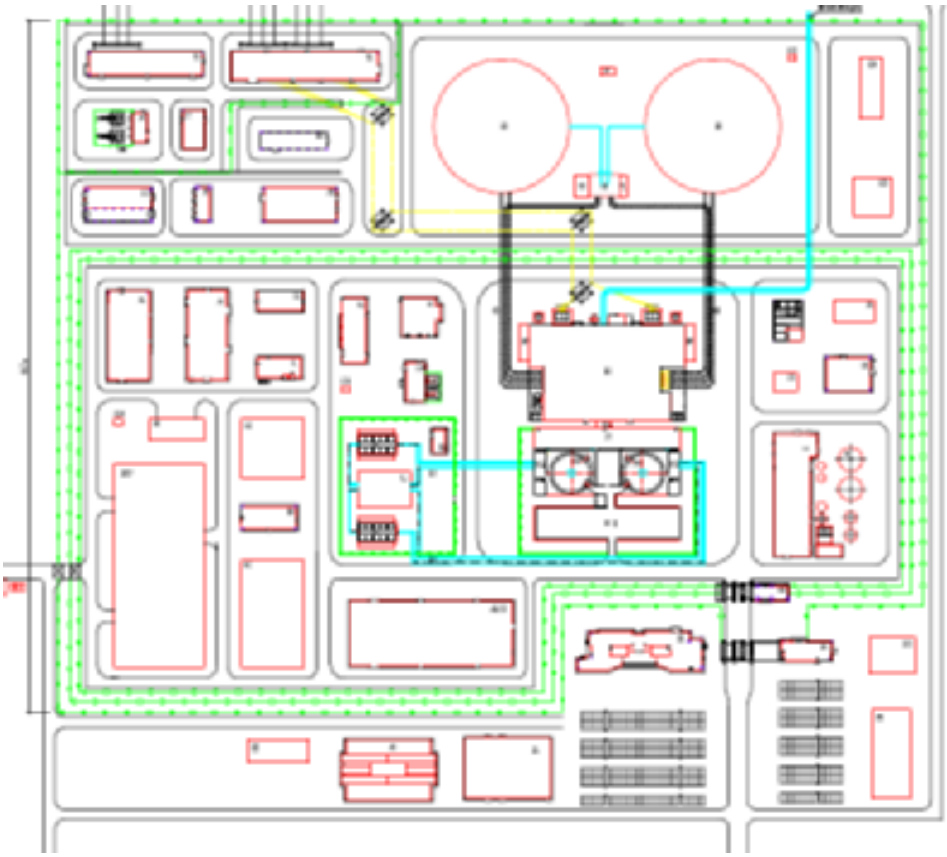


Figure 3: The ACP100 plant layout diagram
(Reproduced courtesy of CNNC)

9. Design and Licensing Status

The ACP100 engineering design is close to completion and a preliminary safety assessment report (PSAR) was recently approved. Passive emergency cooling system, control rod drive

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. CNNC/NPIC has built up comprehensive testing facilities which fulfil the needs of ACP100 design, and the test results on crucial technology provided the necessary basis for final design and safety evaluation of the reactor. Additionally, 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. In April 2015, an agreement to conduct a generic safety review for ACP100 was signed between the IAEA and CNNC. An industrial demonstration plant with two 310 MW(th) unit is planned in Fujian Province, China.

10. Plant Economics

ACP100 is estimated to be competitive against large reactors if more than 4 modules are built simultaneously. In feasibility study, the cost of ACP100 is around 5000 US$ per kW.

CAP150 (SNERDI/SNPTC, China)

1. Introduction

CAP150 is an integral type small modular reactor which employs the most advanced PWR technology, developed by Shanghai Nuclear Engineering Research and Design Institute (SNERDI), a subsidiary of the State Nuclear Power Technology Corporation (SNPTC). It has more simplified systems and more safety than current operating PWRs, to generate an electric power of 150 MW(e*). Figure 1* shows the CAP150 reactor system configuration.

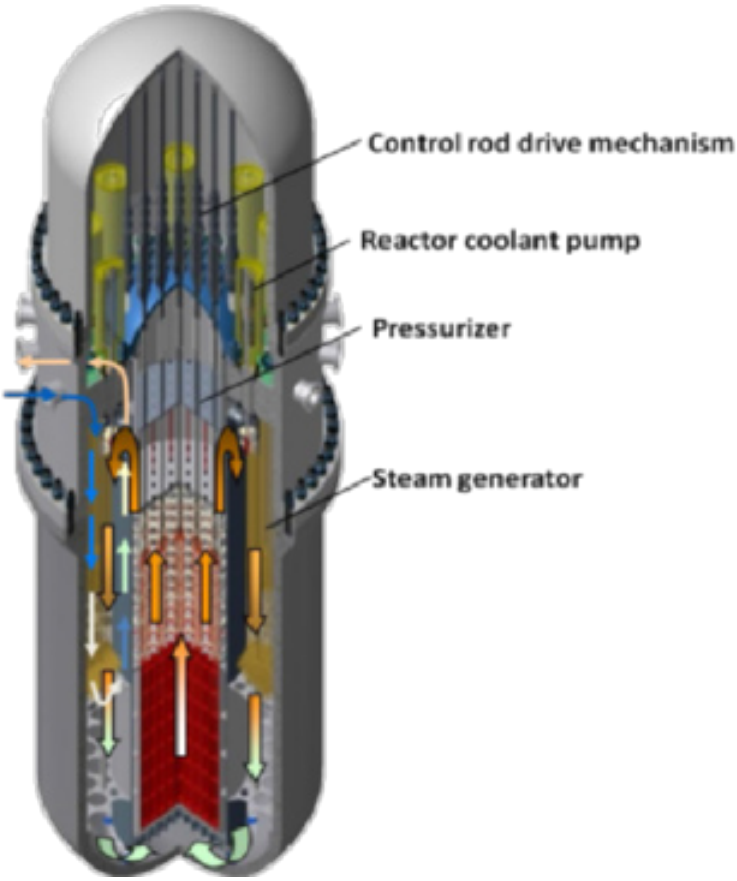


Figure 1: Reactor System Configuration of CAP150
(Reproduced courtesy of SNERDI/SNPTC)

2. Target Application

As a supplement to large reactors, the CAP150 is mainly developed for providing a flexible way for remote electric supply and district heating, and also to replace the old thermal power plants near cities.

3. Development Milestones

Nov 2009	Research work start
May 2010	Conceptual design started
Dec 2013	Conceptual design finished

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer	SNERDI/SNPTC
Country of origin	China
Reactor type	Integral PWR
Electrical capacity (MW(e))	150
Thermal capacity (MW(th))	450
Expected Capacity Factor (%)	95
Design Life (years)	80
Plant Footprint (m^2)	10000
Coolant/moderator	Light water
Primary circulation	Forced circulation
System pressure (MPa)	15.5
Core inlet/exit temperatures (^oC)	290/310
Main Reactivity Control Mechanism	Control rod driving mechanisms and soluble boron
RPV Height (m)	18
RPV Diameter (m)	4.5
Module Weight (metric ton)	<200 (Maximum component weight)
Configuration of Reactor Coolant System	Integrated
Power Conversion Process	Rankine cycle
Fuel Type/Assembly Array	UO_2 pellet/15x15 square
Fuel Assembly Active Length (m)	2.9
Number of Fuel Assemblies	69
Fuel Enrichment (%)	4.5 (average)
Fuel Burnup (GWd/ton)	37 (average)
Fuel Cycle (months)	36
Cogeneration Capability	Possible
Approach to Engineered Safety Systems	Passive
Number of safety trains	4
Refuelling Outage (days)	<20
Distinguishing features	Integrated primary system. Passive safety.
Modules per Plant	2
Target Construction Duration (months)	36
Seismic design (g)	0.30
Predicted core damage frequency per reactor year	< 1E-7
Design Status	Conceptual design finished

4. General Design Description

Design Philosophy

CAP150 design is based on the state of art philosophy of advanced PWR reactor concepts. It aims to innovatively bring out a new G-III+ LWR with higher safety, reasonable and competitive economy, and good engineering feasibility.

The design principle includes:

- Good engineering flexibility, multipurpose.
- Employing proven technology and components to improve the reliability and economy.
- Standardized and modularized Nuclear Steam Supply System (NSSS), the reactor coolant system (RCS) modules are fully manufactured in factory and transported to site for installation.
- Defence in depth. Diverse safety systems include active and passive features. The ultimate heat sink of passive long term cooling is air.
- Enhance safety features of spent fuel storage.
- Environment and public friendly, eliminating offsite emergency requirements.
- The probabilistic safety analysis (PSA) and deterministic analysis are performed simultaneously to achieve the safety goals from the early stage of design.

Nuclear Steam Supply System

CAP150 employs integral design, main equipment in the RCS including the pressurizer (PZR), steam generators (SGs), reactor coolant pumps (RCPs) and control rod drive mechanism (CRDM) are all integrated into the reactor pressure vessel (RPV). Main pipelines are eliminated. The RPV is 4.5 m in diameter and about 18 m in height.

Fuel and Reactor Core

Proven fuel assemblies of 300 MW(e)-class NPPs are used in this SMR. The FA300 fuel assembly is developed by SNERDI, and it has been widely used in Qinshan NPP(in China) and Chashma NPP(in Pakistan), presenting very good mechanical and physical behaviour. *Figure 2* shows the schematic diagram of the fuel. The FA300 is suitable for a lower burnup feature of SMR, it is currently available and good to shorten the development and reduce R&D cost.

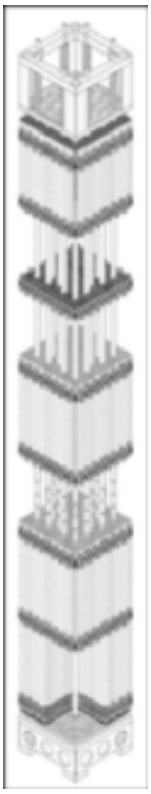


Figure 2: Fuel Assembly of CAP150
(Reproduced courtesy of SNERDI/SNPTC)

There are totally 69 FAs in the core with 9x9 arrangements. The height and diameter of active core is 2.9 m and 2m, respectively. Steel-water reflector is employed to improve the neutron economy and reduce the irradiation damage to vessel.

Reactivity Control

There is no soluble boron in the core of CAP150. Instead, only control rods are used to control the reactivity. The boron-free core eliminates systems for boron adjustment and PH value control, simplifies the operation complexity and frequency of chemical control system, greatly reduces the release of liquid radwaste, and avoids boron dilution accidents. The design of boron-free core accords with requirements for small reactors like system simplification, easy operation and low radioactivity. Another advantage of the mechanical shim control is its quick transient responses and strong ability of load following. This makes the SMR suitable for operating at variable loads, which is especially important for a small size grid or a region with only single power supply. Electromagnetic CRDMs are installed on the inner head of the RPV, and connected with control rod assemblies including adjustment assemblies and safety assemblies.

Reactor Pressure Vessel and Internals

The RPV is assembled in three parts (upper, mid, lower), connected with flange joint. The SG, PZR, RCP, CRDM are all installed in the mid-part. There are no penetrations on the upper and lower-part of RPV. The pressure boundary of the upper and lower RPV is simple and safe. Because the RCP's motor and CRDM are not directly in contact with the coolant, these components will not suffer high temperature and high radiation. The RCP and CRDM are based on current technology, and also possess higher reliability and maintainability. There are double heads at the upper part of RPV. Two physical barriers ensure the integrity of pressure boundary, and prevent coolant loss and radioactive release. Pressure is pre-charged by high pressure nitrogen between the inner and outer head so as to reduce the probability and consequence of rod ejection accidents. In addition, the pre-charged pressure can also eliminate the probability of the any leakage and break of the many pipelines and penetrations on the annular support platform and also the inner head. *Figure 3* shows the schematic diagram of integral pressure vessel and internals.

Reactor Coolant System

The primary coolant is heated and flows upwards in the core, goes through the riser, and then from the upper parts enters the annular down-comer outside the barrel. Eight SGs are housed in the upper part of the down comer. The coolant is pumped downwards through the SGs and cooled down and then fed back into the core again.

Steam Generator

There are eight SGs, which are circularly distributed in the RPV down comer. The inlet of feed water and outlet of steam are located on the annular support platform. The in service inspection of the tubes to monitor failures or blocking are performed from the secondary side. Each SG corresponds to one RCP. One of eight SGs failure will not cause significant impact on normal operation. The design pressure of steam/feedwater pipelines is the same as that of the primary loop. Accidents such as steam generator tube rupture (SGTR) and steam line break (SLB) can be quickly mitigated by closing SG isolation valves. In addition, SGs are also used as heat exchangers of secondary side residual heat removal system.

Pressurizer

The pressurizer is located on the top of the riser. The CRDMs pass through the upper dome of

the pressurizer. The saturated fluid in the pressurizer body is connected with the subcooled coolant in the riser through surge holes setting in the bottom plate of the sleeve. Adjustment of pressure in the pressurizer is achieved by heating and spraying, to control the pressure in the primary loop.

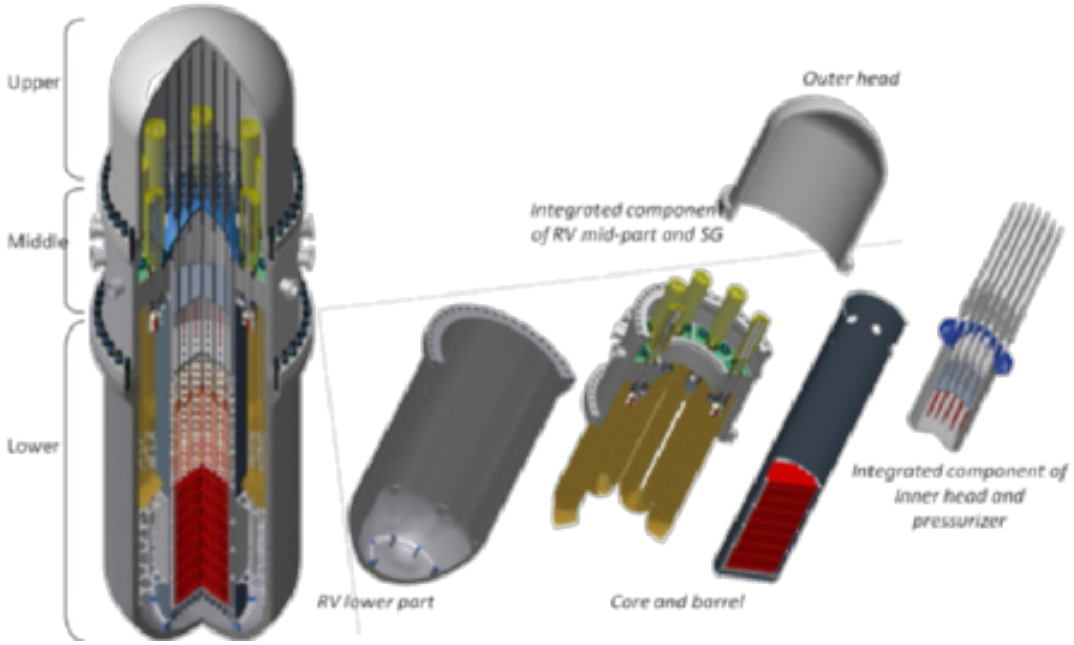


Figure 3: Integrated RCS and components
(Reproduced courtesy of SNERDI/SNPTC)

5. Safety Features

CAP150 has some inherent safety features, as shown in the following aspects:

- RCS adopts integral design, which eliminates large pipelines and reduces small size pipelines. This design basically eliminates large break LOCAs and reduces the occurrence frequency of small break LOCAs.
- Double heads of the RPV further increase the integrity of pressure boundary in the primary loop and radioactivity barrier.
- Separated configuration of multiple sets of RCPs and SGs decreases the impact of SLB and rotor seizure when one set of these equipment failure.
- The design pressure of steam/feedwater pipelines are the same as that of the primary loop. Accidents such as SGTR and SLB can be quickly mitigated by closing isolation valves.
- Small flow resistances in the integral reactor, strong natural circulation ability, small core power and low power density are helpful for the mitigation of accidents such as station blackout and loss of flow accident.
- Large water inventory and large thermal inertia ensure enough response time for accident mitigation.
- There is no soluble boron in the coolant, and thus the core has a better negative feedback coefficient.
- The steel containment is submerged in containment cooling pool. If some leakage occurs, the radioactivity retention by containment cooling pool and concrete can further reduce the release of radioactivity to environment.

- The in-containment spent fuel pool assures continuous cooling of spent fuel after accidents as well as containment of high radioactivity.

Engineered Safety System Approach and Configuration

in addition to the inherent safety features, engineered safety systems are considered to enhance the safety. Both active and passive systems are employed to contribute a more diverse approach and decay heat removal paths. *Figure 4* shows some passive safety systems in the CAP150, including the passive decay heat removal by SG secondary side, passive decay heat removal by primary side, simplified passive safety injection system, and passive containment cooling system.

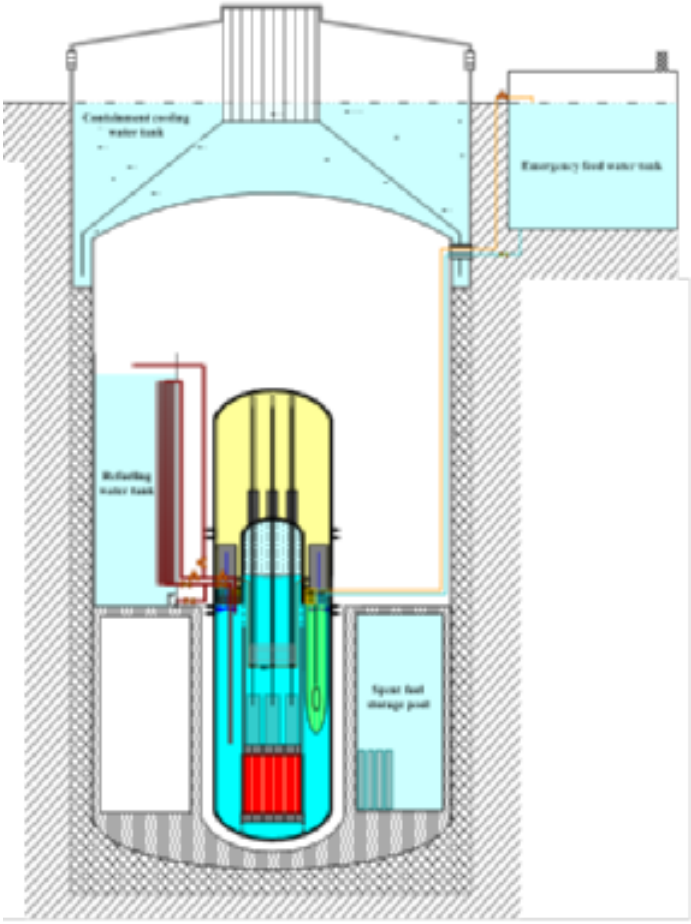


Figure 4: Passive safety system of CAP150
(Reproduced courtesy of SNERDI/SNPTC)

Decay heat removal system

In normal operation, the reactor heat is removed by secondary loop. Feedwater system is non-safety-related. It can be taken as the first heat removal channel to maintain the cooling of the core. There are totally 8 SGs and only two of them are needed for coolant cooling after accidents. The normal residual heat removal system can be put into use in both normal operations and accident conditions. Two sets of the system are deployed. Each of them has 100% capability to remove core decay heat. Its pipelines and heat exchangers are designed to endure the primary pressure. Thus it can be used to remove the residual heat in emergency conditions. The system is safety-related and can be taken as the secondary heat removal channel to cool the reactor. Both passive residual heat removal systems in the primary loop and that in the secondary side are safety-related systems. These two systems may be initiated

at the same time by safety signals. They may be taken as the third heat removal channel.

Emergency core cooling system

The safety injection system of the CAP150 is greatly simplified compared with operational PWRs. In current PWR power plant, the safety injection system has three main functions. The first is to provide emergent and durative cooling for the core when LOCA occur. The second is to compensate the reactor coolant inventory for small leaks or depressurizations to maintain water level in pressurizer while chemical and volume control system (CVCS) is failed. The third is to inject high concentrated boron solution into the core so as to prevent the core from re-criticality when over cooling events such as SLB takes place. In order to achieve these three functions, current safety injection system has three subsystems: high pressure injection, accumulator and low pressure injection. The high pressure injection implements the second function while CVCS is invalid and the third function when SLB happens. When LOCA takes place, the flow offered by high pressure injection is too low to maintain core covered. Accumulator is then needed to provide a very high flow rate injection at a middle pressure level. As the system depressurizes to a pressure low enough, the low pressure injection starts and remove decay heat for a comparative long term. The design features of CAP150 make it possible to greatly simplify its safety injection system. The design pressure of the normal residual heat removal system is the same as that of the primary loop. The second function of the safety injection system can be performed by the normal residual heat removal system. Since the seriousness of the SLB accident decreases, the optimized design of control rod assembly makes the core shutdown margin enough and core retune to critical impossible. Thus the third function of the safety injection becomes unnecessary. The only requirement left for safety injection system of CAP150 is to mitigate small break LOCA. For CAP150, large break LOCA is eliminated because of integrated RCS so that accumulator injection is not quite necessary. The core power is limited and the power density is quite low. The specific coolant inventory is very large compared to current PWRs. The containment free volume is small enough that the pressure between reactor vessel and containment can reach balance very soon after small break LOCA. Even if there is no high pressure injection, the core will not uncover before low pressure injection starts. Hence for CAP150, it is believed that only low pressure injection is needed to maintain core safety. Therefore, only the low pressure injection system is deployed in CAP150. As shown in *Figure 4*, it consists of a refuelling water tank, auxiliary depressurization system and the related valves and pipelines.

Containment system

The upper part of the containment is submerged in the containment cooling water pool. When LOCA or SLB accident occurs, steam blows into the containment, steam condenses on the wall of the steel containment, and the heat is transferred to the containment cooling pool. The condensed water partly returns to the refuelling tank for the low pressure safety injection, and partly flows into the sump. The water in the containment cooling pool will be evaporated and the water inventory can last 7 days without any human intervention. In addition, there is a supplement pool to make up for the containment cooling pool when its water level becomes low so as to keep the passive cooling of the containment for a long term. This containment cooling system doesn't need actuation signals and can perform its safety functions as soon as accidents occur. It runs fully by natural circulation. The principle of the system is simple and reliable. Moreover, the containment cooling pool provides an additional barrier for radioactivity and thus further decreases release possibility. The spent fuel storage pool is located in the containment, so that the safety of spent fuel under SBO is ensured by the passive containment cooling systems. And all radioactivity is contained in the containment during normal operation or an accident.

6. Plant safety and Operational Performances

Using small event tree / large fault tree methodology, preliminary level one PSA analysis of internal events at power for CAP150 was carried out. Based on analyses and postulations on initiating events, event tree models and success criteria, corresponding model was built including initiating events frequency, event tree models, fault tree inputs, human error and corresponding processing and so on. Preliminary quantified analysis results show that the core damage frequency (CDF) is about 3.5×10^{-8}/reactor-year. LOCA events are the largest contribution accounting for about 40%. It is estimated that the large release frequency (LRF) of CAP150 is about 10^{-9}/reactor-year in magnitude or even less because of several design features such as the retention of core debris and passive containment cooling are considered, the core power level is low, the time without human intervention is relatively long, the whole containment is submerged in the containment pool so that one more barrier against the large radioactivity release is added etc.

7. Design and Licensing Status

Conceptual design completed.

CAP200 (SNERDI/SNPTC, China)

1. Introduction

The China Advanced Passive pressurized water reactor 200MW(e) (CAP200) is one of the serial research and development products of PWRs adopting passive engineered safety features initiated by SNERDI. The design of CAP200 is based on the experience of the PWR technology R & D for more than 45 years, construction and safe operation for more than 20 years in China. It is the outcome of accumulated experience and achievements of the world's first batch of AP1000 units and the R&D of CAP1400. Furthermore, it adopts safety enhancement measures based on lessons learnt from Japanese Fukushima nuclear accident. *Figure 1* shows reactor system configuration of CAP200.

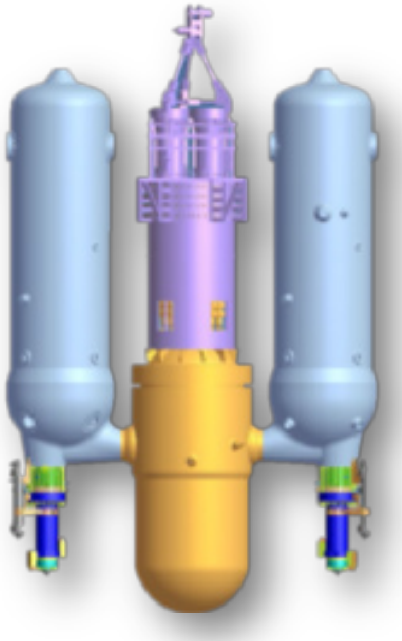


Figure 1: Reactor System Configuration of CAP200
(Reproduced courtesy of SNERDI/SNPTC)

2. Target Application

CAP200 can be used as a supplement to large PWRs, this reactor is designed for multiple applications, such as nuclear cogeneration and replacing retired fossil power plants in urban areas.

3. Development Milestones

May 2014	Conceptual design started
December 2015	Conceptual design finished

4. General Design Description

Design Philosophy

CAP200 is a small PWR which is designed with improved safety, flexibility and environmental friendliness, and is also comparable with other SMRs on economy. Compared

with large PWRs, CAP200 has a number of advantages such as higher inherent safety, lower frequency of large radioactivity release, longer time without operator intervention, smaller environmental impact, lower site restrictions, shorter construction period and smaller financing scale as well as lower financial risk. The main features of this reactor are as follows:

- Compact layout primary system: Steam generators (SG) are connected to reactor pressure vessel (RPV) directly and main pipes are eliminated. Compact layout of system and components results in lower risk and possibility of loss of coolant accidents and smaller primary system footprint.
- Modularized: main modules can be fabricated in factory and transported to the site for installation. Construction period can be shortened because of high modularization and the use of existing infrastructure.
- Redundant and diversified passive safety features: redundant and diversified active and passive safety features are deployed, which ensures the reactor core safety and extremely low risk of large radioactivity release.
- Steel containment is located below ground level, which eliminates the risk of outside attacks and improves the ability of preventing natural disasters.

Primary circuit and its main characteristics

CAP200 has two primary coolant loops which transfer heat generated by fission reaction of U_{235} in reactor core to SG. For each loop, it consists of one SG, one main coolant pump and pressure nozzles connecting them. A pressurizer (PZR) is connected to reactor coolant system by surge line. The SG is located at higher elevation than RPV in order to achieve natural circulation. The elevation of PZR is higher than that of pressure nozzles which enhances the drainage of PZR liquid to the reactor coolant system and reduces the carry-over of liquid to PZR when the depressurization system is activated. The design of surge line minimizes the potential for thermal stratification. As the barrier against release of radioactivity, the reactor coolant system pressure boundary is designed to provide a high degree of integrity for operation of the plant. CAP200 employs compact design, which means that main equipment in the reactor coolant system (RCS) including the RPV, SGs and reactor coolant pumps (RCP) are connected by short pressure nozzles so that main pipelines are eliminated. RPV and SG are connected by a straight nozzle and SG and RCP are connected by an elbow. In each pressure nozzle an inner duct is cased. Hot water leaving reactor core flows through the inner circular duct into hot channel of SG. After heat release in the SG heat transfer tubes, cold water flows from cold channel of SG into pump suction through inner elbow duct and is pumped through the outer annular elbow back into third chamber of SG and then into downcomer of RPV through outer annular duct.

Reactor coolant pump

Both leak-tight canned motor pump or wet coil pump are possible choices for the RCP of CAP200. There is rich experience of use of canned motor pump and wet coil pump in nuclear power plants, and the current advanced large nuclear reactors are also employing canned motor pump or wet coil pump. The RCP is designed to produce a head of 65m at design flow rate of 12000 m3/h with a cold leg temperature of 289°C. The reactor coolant pump has no shaft seals, eliminating the potential for seal failure LOCA, which significantly enhances safety and reduces pump maintenance. The pumps have an internal flywheel to increase the pump rotating inertia and thereby providing a slower rate-of-flow coastdown to improve core thermal margins following the loss of electric power.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer	SNERDI/SNPTC
Country of origin	China
Reactor type	PWR
Electrical capacity (MW(e))	>200
Thermal capacity (MW(th))	660
Expected Capacity Factor (%)	>93
Design Life (years)	60
Coolant/moderator	Light water
Primary circulation	Forced circulation
System pressure (MPa)	15.5
Core inlet/exit temperatures (°C)	289/313
Main Reactivity Control Mechanism	Control rod drive mechanisms and soluble boron
RPV Height (m)	8.545
RPV inner Diameter (m)	3.280
Module Weight (metric ton)	<200 (Maximum component weight)
Configuration of Reactor Coolant System	Compact
Power Conversion Process	Rankine cycle
Fuel Type/Assembly Array	UO_2 pellet/17x17 square
Fuel Assembly Active Length (m)	2.4
Number of Fuel Assemblies	89
Fuel Enrichment (%)	4.2 (average)
Fuel Burnup (GWd/ton)	37 (average)
Fuel Cycle (months)	24
Cogeneration Capability	Possible
Approach to Engineered Safety Systems	Passive
Number of safety trains	2
Refuelling Outage (days)	<20
Distinguishing features	Compact layout; Passive safety; Underground containment.
Modules per Plant	1
Target Construction Duration (months)	36
Seismic design (g)	0.3
Predicted core damage frequency per reactor year	<1E-6
Design Status	Conceptual design finished

Reactor core and fuel design

The reactor core and fuel serves as the source of heat generation in the reactor coolant system. The reactor core consists of fuel assemblies, rod cluster control assemblies, grey rod cluster assemblies, burnable absorbers, neutron sources, and thimble plugs. The fuel assembly is typical 17×17 lattice type used in commercial PWR, with shorter length and part of the grids number reduced. The assembly structure consists of cladding material, and UO_2 ceramic pellet, with an enrichment of less than 5%, complying with regulation rules for civil nuclear facility. IFBA burnable poison is present in the assembly to suppress the initial excess reactivity. The core rated thermal power is 660 MW(th) with 89 fuel assemblies. The

core has an equivalent diameter of 2.29m, and the total uranium load is about 26t. High burnup and low neutron leakage reactor core fuel management strategy has been used for CAP200. For the advanced first core design, there are three radial regions that have different enrichments ranging from 2.00 to 4.00% U^{235} to enhance the fuel economics. The active fuel length of reactor core is 240 cm. The linear averaged power density and volume averaged power density for CAP200 are 11.4kW/m and 66.9MW/m3, respectively. The lower power density leads to lower fuel center temperature, both in normal operation and transients. As a result, the core has a good intrinsic safety feature with regard to the fuel. The core is designed for a fuel cycle of 24 months with 93% availability, and averaged discharge burnup of assembly is as high as 42218MWd/tU for balance cycle. CAP200 has at least 15% margin for departure from nucleate boiling (DNB). The power reactivity coefficients, including fuel Doppler, moderator temperature and moderator void coefficients are negative providing negative reactivity feedback characteristics and meeting the design criteria, which enhances safety of reactor core inherently. CAP200 have enough margins for several operation modes with the maximum value control rod withdrawal. The peak power factor of CAP200 satisfies the limiting value for every fuel cycle. Core reactivity is controlled by both soluble boron and control rods. CAP200 is capable of load following without boron dilution. Nevertheless, the control rods need repositioning when boron dilutes. This method suppresses the excess reactivity with soluble boron as conventional PWR does, but to a large degree simplifies the conventional boron system and the dilution operation. The CAP200 reactor core adopts two different types of control rod assemblies. There are 37 assemblies in total. The high worth assemblies are known as “black” rods. These are used for shutdown and large swings of reactivity. The grey rods with lower worth are used for load-follow during operation of a fuel cycle to avoid adjusting soluble boron concentration, which results in a substantial reduction in waste water generation and treatment for PWR required to execute load-follow operation.

Reactor pressure vessel

The reactor pressure vessel (RPV) of CAP200 is designed with reference to that of CAP1400. The difference is in the inlet and outlet of RPV. The pressure nozzles are used in the case of CAP200, to connect RPV and SGs replacing main pipes. The RPV is a high-pressure boundary used to support and enclose the reactor core. The vessel is cylindrical with a removable flanged hemispherical upper head and hemispherical bottom head. The height of RPV is 8.545m and inner diameter of reactor core region is 3.280m. The area with highest neutron fluence of active core region of the vessel is completely forged and free of welding which enhances the confidence of 60 years’ design life and decreases the task of in service inspection. As a safety enhancement, neutron and temperature detectors enter the reactor through upper head of reactor vessel which eliminates penetrations in the lower head of RPV. This reduces the possibility of a loss of coolant accident (LOCA) and uncovering of the core The core is positioned as low as possible in the vessel to decrease re-flood time in the event of an accident; furthermore, this arrangement is very helpful for successful execution of IVR.

Reactor internals

The reactor internals, installed in RPV, provide support, protection, alignment and position for the core and control rods to guarantee safe and reliable operation of reactor. The reactor internals consist of the lower internals and the upper internals. The core shroud made of stainless steel is welded which eliminates the occurrence of reactor core damage induced by loosening of baffle bolts. The reactor internals design of CAP200 takes the inner pressure nozzle installation into consideration.

Integrated head package

The integrated head package (IHP) is installed on the top of the reactor vessel head, and

combines several components into one assembly to simplify refuelling operation and decrease the radiation dose for refuelling and maintenance personnel. The IHP design of CAP1400 renders a reference for that of CAP200. The CRDM cooling system, including ventilation panels, fans and its supports, usually used in the conventional PWRs are removed for CAP200 since the high temperature performance of coil in CRDM of CAP200 can withstand the temperature of 350℃. The structure of IHP of CAP200 is much simpler. It includes shroud assembly, lift rig assembly, radial arm hoist assembly, CRDM seismic support, cable and cable support structure.

Steam generators

The steam generator (SG) of CAP200 is vertical U-tube type with 3861 tubes made of thermally treated nickel-chromium-iron Alloy 690 on a triangular pitch, to fulfil the requirement of heat transfer capacity. The design parameter of steam exit from SG is 6.02 MPa and 183.2 kg/s for thermal design flow rate with no tube plugged, respectively. Pressure nozzles are used to connect RPV and RCP directly with SG. The inner duct of the pressure nozzle connecting RPV with SG side is welded to the partition board of SG water chamber. The nozzles are designed with two ducts. For the nozzles between RPV and SGs, hot water flows in the inner circular duct and cold water in the outer annular duct. Stainless steel bars have been installed in bend area to preclude occurrence of damaging flow induced vibration under all conditions of operation. The feedwater spray nozzle is at top of the feed ring and adopts the separated start-up feedwater pipe which eliminates thermal stratification and prevents occurrence of water hammer. The SG channel head is divided into three parts, hot channel, cold channel, and a third channel through which the coolant is pumped back into reactor vessel.

Pressurizer

The pressurizer of CAP200 is a typical steam-type one with electrical heater at the bottom and spray at the top. The pressure control is steady and reliable. The pressurizer is a vertically mounted cylindrical pressure vessel with hemispherical top and bottom heads which adopts the traditional design that is based on proven technology. The pressurizer volume is designed to be large which increases margins for transient operation and minimizes the number of unplanned reactor trips and provides higher reliability. In addition, fast-acting power-operated relief valve, which is one of the reactor coolant system leakage sources and the component requiring potential maintenance, won't be needed because of the improved transient response by using large volume pressurizer.

Control Rod Drive Mechanism

The newly developed control rod drive mechanism (CRDM) of CAP200 will be able to operate the motor and coil assembly at higher temperature. The heat generation and heat transfer coefficient of the CRDM coils are improved to reduce the cooling air flow rate, which will also contribute to the simplification of the IHP since the cooling fan and related seismic loads can be reduced.

5. Description of safety concept

Safety requirements and design philosophy

The fundamental design philosophy of CAP200 is defence in depth including accident resistance, core damage prevention and mitigation. Accident prevention is realized by using the best available material and water chemistry, high quality requirement for components during design and construction, negative reactivity coefficients, improved man-machine interface system and extended operator response time. In addition, the need for offsite

emergency response shall be eliminated. For core damage prevention, it is mainly achieved by passive safety features which are adequate to automatically establish and maintain core cooling and containment integrity for a significant period of time after initiation of design basis events assuming the most limiting single failure, no operator action, and with no onsite and offsite ac electrical power sources. In order to mitigate accident, containment systems are provided for heat removal and retention of fission products. Hydrogen control system is provided to achieve that the hydrogen concentration in containment does not exceed 10% under dry conditions for an amount of hydrogen equivalent to that generated by oxidation of 100% of the active fuel clad. In addition, redundancy has been used for containment components, such as containment isolation valve. For CAP200, defence in depth is also realized by some non-safety systems which decrease the events that may lead to core damage. The core damage frequency and large release frequency are aimed at lower than 1.00×10^{-6} and 1.00×10^{-7}, respectively.

Safety systems and features

CAP200 adopts passive safety systems which take advantage of natural forces such as natural circulation, gravity and compressed air to make the systems work, offering improvements for plant in simplicity, safety, O&M, availability and investment protection. No active components such as pumps, fans and other machinery are used. A few simple valves align and automatically actuate the passive safety systems. The passive safety systems are designed to meet criteria of single failure, independence, diversity, multiplicity. For CAP200, passive core cooling system and passive containment cooling system are adopted.

Passive containment cooling system

The containment of CAP200 is submerged in a water pool. After a steam line break accident or a loss of coolant accident, heat will be transferred from steam in containment to the water pool. The water pool is safety-related and prevents the containment from exceeding the design pressure and temperature following a postulated design basis accident by cooling the outside surface of containment, as shown in *Figure 2*. The inventory in the water pool can last at least 7 days after an accident. The Passive containment cooling system works without operator control or external assistance. The passive containment cooling system performs the following safety-related functions:

- Reduce the containment temperature and pressure following a LOCA or main steam line break (MSLB) accident inside the containment by removing thermal energy from the containment atmosphere.
- Retain fission product by gravitational deposition, thermophoresis and condensation to wall of containment steel, all of the working mechanism are passive.
- Transfer heat to the safety-related ultimate heat sink for other events resulting in a significant increase in containment pressure and temperature.
- Limit releases of radioactivity (post-accident) by reducing the pressure differential between the containment atmosphere and the external environment, thereby diminishing the driving force for leakage of fission products from containment to atmosphere.
- Provide a seismic source of makeup water to the spent fuel pool in the event of a prolonged loss of normal spent fuel pool cooling and provides a seismic source of makeup water to the fire protection system.

Passive core cooling system

The main function of passive core cooling system is to provide emergency core cooling during postulated design basis accidents by supplement and boration to RCS after non-LOCA accidents and safety injection to core after LOCA. Passive core cooling system forms core decay heat removal pathway together with passive containment cooling system. For CAP200, Passive Safety Injection System combined with the Passive Decay Heat Removal System is referred as the Passive Core Cooling System. The Passive Safety Injection System consists of two core makeup tanks, one in containment refueling water storage tank (IRWST), IRWST screen, containment recirculation screen, pH adjustment basket and associated valves, piping, instrumentation etc., as shown in *Figure 2*. The passive decay heat removal system consists of one passive residual heat removal heat exchanger and associated valves, piping, and instrumentation The heat exchanger is located in the IRWST, which works as the heat sink for the heat exchanger.

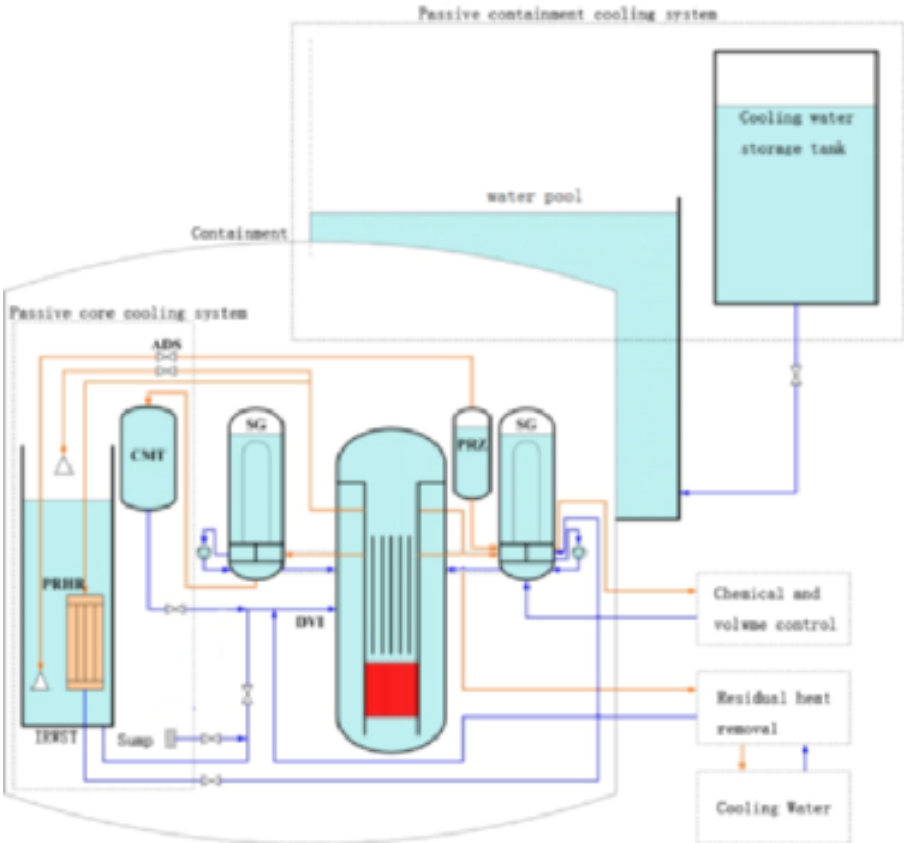


Figure 2: Passive Core Cooling System and Passive Containment Cooling System (Reproduced courtesy of SNERDI/SNPTC)

Long-term accident mitigation

The passive containment cooling system will remove the heat from containment within 7 days after the initiation of a postulated accident and keep the temperature and pressure without exceeding the design values assuming no operator action and on-site or off-site ac power. Within 7days, the inventory from the water pool guarantees that the containment pressure won't exceed the design value. After that the decay heat is low enough for the heat pipes to transfer decay heat from the water pool to atmosphere and keep the water pool level not decreasing. In addition, supplement to the water pool can be obtained from a cooling water storage tank. The non-safety related cooling water storage tank and the heat pipes keep the containment cooled for indefinite duration.

Hydrogen control system
During degraded core accident, hydrogen will be generated at a greater rate than during the design basis LOCA, and the hydrogen control system is designed to deal with the risk. The hydrogen control system consists of hydrogen monitoring system and Passive Auto-catalytic Recombiners (PARs) system. The hydrogen detectors are installed in the region of containment dome to measure the concentration of hydrogen which may be transferred to main control room and remote shutdown station. Passive auto-catalytic recombiner system is installed to control the hydrogen concentration in containment without exceeding regulatory requirement.

In-vessel retention of molten core debris
The IVR of molten core debris is one of the key mitigation strategies during severe accidents for CAP200. During postulated severe accidents, flooding the reactor cavity with the injection of IRWST water to submerge the RPV will be performed to implement the IVR strategy and to maintain the integrity of RPV by cooling the external surface of the vessel. This strategy prevents molten debris from relocating to the containment which may cause further ex-vessel severe accident phenomena such as ex-vessel steam explosion and molten core-concrete interaction.

Lessons learned from Fukushima nuclear accident
The resistance capacity of CAP200 to extreme natural hazards such as earthquake, flood and etc. has been enhanced. Since the containment is buried underground, CAP200 can also withstand the impact of aircraft crash. Especially, without being dependent on alternating current, the passive safety system is able to keep CAP200 safe within 7 days after accident initiation; After 7 days, the non-safety grade measures are available to offer reactor core cooling for indefinite duration; In addition, the reactor core can still be cooled with some extra off-site assistance.

6. Nuclear island layout

The size of Nuclear Island is minimized by both system simplification and adoption of passive design. The number of system component is reduced by adoption of large capacity equipment, common use of single equipment for different systems.

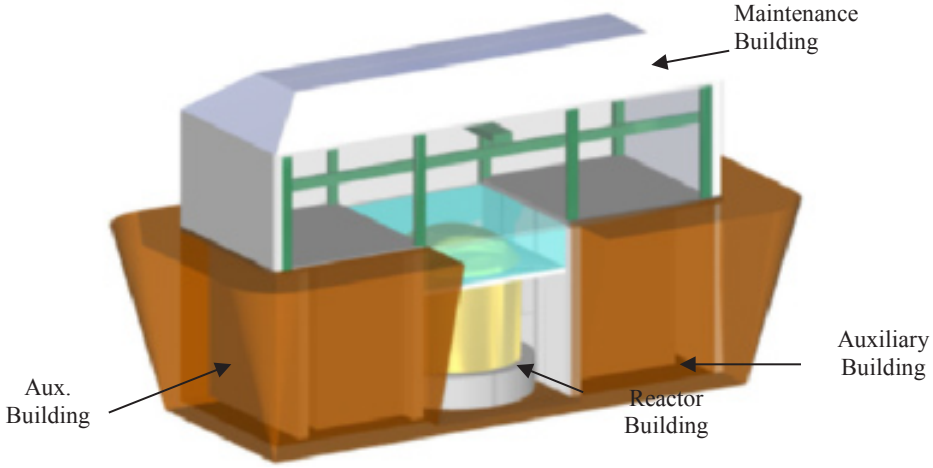


Figure 3: Layout of Nuclear Island (Reproduced courtesy of SNERDI/SNPTC)

7. Design and Licensing Status

Conceptual design finished.

AHWR-300 (BARC, India)

1. Introduction

The Indian Advanced Heavy Water Reactor (AHWR) is a vertical pressure tube type light water cooled and heavy water moderated reactor to generate 300 MW(e). The reactor concept adopts the proven pressurized heavy water reactor (PHWR) specific technologies pertaining to pressure tube and low pressure moderator based design. The reactor is designed and developed by Bhabha Atomic Research Centre (BARC). The AHWR aims to produce most of its power from thorium, utilizing large-scale thorium for future commercial nuclear power plants. The shop assembled coolant channels of the AHWR enables quick replacement of pressure tube without affecting other installed coolant channel components. The layout diagram of the AHWR plant is depicted in *Figure 1*. The reactor incorporates several advanced passive safety features. The AHWR is designed to provide protection against external as well as internal security threats, including insider malevolent acts. The plant design contains the following features:

- Heat removal from core under all operating conditions including shutdown condition by natural circulation of coolant;
- Passive containment cooling (PCC) and containment isolation during loss of coolant accident (LOCA);
- Passive removal of decay heat of reactor in case of station blackout 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;
- 100-year target design plant life;
- Direct injection of cooling water by emergency core cooling system (ECCS) in a passive mode directly inside fuel cluster;
- Passive poison injection into the moderator in the event of non-availability of both the shutdown systems due to wired system failure or malevolent action;
- Passive cooling of moderator and endshield in case of station blackout;
- Dedicated core catcher to cool the core melt in case of very low probable severe accident;
- Hardened filtered vent system to depressurize the containment and retain radioactive nuclides in case of severe accident;
- A design objective requiring no exclusion zone beyond plant boundary on account of its advanced safety features;

2. Target Applications

The AHWR is designed for electricity generation as well as sea water desalination, with a target to produce 2400 m^3/day of demineralised water. It can also be configured to deliver higher desalination capacities with some reduction in electricity generation.

3. Development Milestones

2003 September	Peer review of the design performed
2006	Design review completed

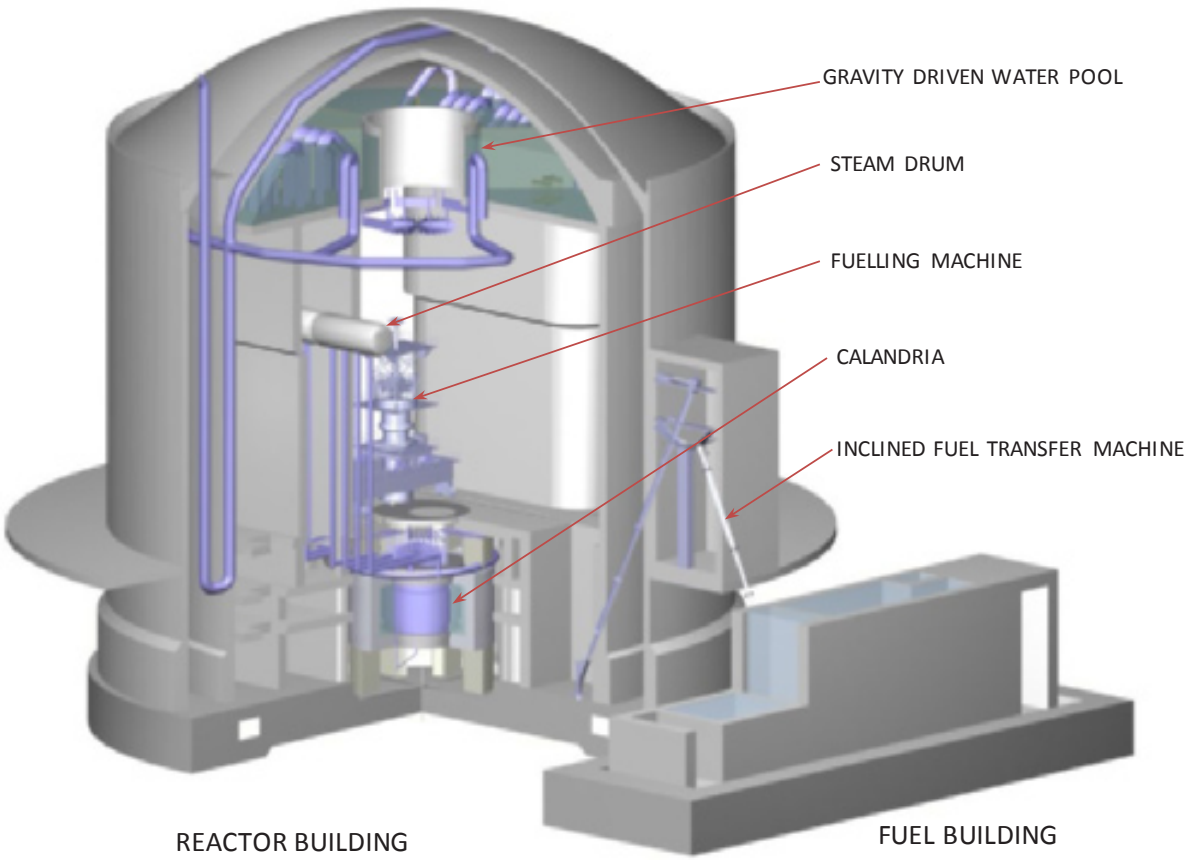


Figure 1: Layout diagram of an AHWR plant (Reproduced courtesy of BARC)

4. General design description

Design philosophy

The AHWR is designed and developed for large-scale use of thorium in commercial nuclear power generation with no external input of U_{233} in the equilibrium cycle. The AHWR incorporates inherent and passive safety features to the maximum extent as a part of the defence in depth strategy. The AHWR design aims to establish a case for elimination of the need for evacuation planning in case of accident scenario. The AHWR passive and active hybrid safety systems designed to cope with the consequences of design basis accidents. The AHWR is also designed for seawater desalination. The AHWR schematic diagram is depicted in *Figure 2*.

Nuclear steam supply system

The nuclear steam supply system (NSSS) consists of the reactor core and internals, coolant channels, tail pipes and steam drums. The main heat transport (MHT) system transports heat from fuel pins to the steam drums, using light water as the coolant under natural circulation. The MHT system consists of a common circular inlet header from which feeders branch out to the coolant channels in the core. The outlets from the coolant channels are connected to tailpipes carrying steam-water mixture from the individual coolant channels to the four (4) steam drums.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer	Bhabha Atomic Research Centre (BARC)
Country of origin	India
Reactor type	Pressure tube type heavy water moderated reactor(vertical flow)
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
Core inlet/exit temperatures (^oC)	258.2 / 285 with average steam exit quality of 19.7%
Main Reactivity Control Mechanism	Rod insertion
Calandria Height (m)	5
Calandria Diameter (m)	6.9
Calandria Weight (ton)	50
Configuration of Reactor Coolant System	Integrated
Power Conversion Process	Rankine Cycle
Fuel Type/Assembly Array	Flexibility in fuel options: Can use Either (Pu-Th)MOX and (Th-U_{233})MOX or ((Th-LEU) MOX with LEU of 19.75% U_{235} enrichment/54 pins (12, 18, 24)
Fuel Assembly Active Length (m):	3.5
Number of Fuel Assemblies:	452
Fuel Enrichment (%):	4.21% average fissile content
Fuel Burnup (GWd/ton):	60
Fuel Cycle:	Th-Pu-U233 or LEU - Thorium
Cogeneration Capability:	Possible
Approach to Engineered Safety Systems	Passive
Number of safety trains:	4
Refueling Outage (days):	On-power
Distinguishing features:	Natural circulation driven heat removal during normal operation and hot shutdown, Passive containment cooling and isolation system during LOCA, Passive corium cooling by core catcher, Double containment system. Passive moderator and end shield cooling during SBO. Passive decay heat removal using ICs in case of SBO
Modules per Plant:	1
Target Construction Duration(months):	~60
Seismic design (g):	0.20
Predicted core damage frequency (per reactor year):	~ 1 E-8
Design Status:	Basic design completed

Reactor core

The reactor core is housed in calandria, a cylindrical stainless steel vessel containing heavy water which acts as a moderator and reflector. The core consists of a total 513 lattice

locations arranged in square pitch of 225 mm. There are452 coolant channel assemblies in the core, and it encloses the pressure tubes that are arranged in a vertical orientation to facilitate natural circulation flow. The circular fuel cluster of the AHWR shown in *Figure 3* contains 30 (Th, U_{233} MOX pins and 24 (Th, Pu) MOX pins, along with a displacer rod at the centre.

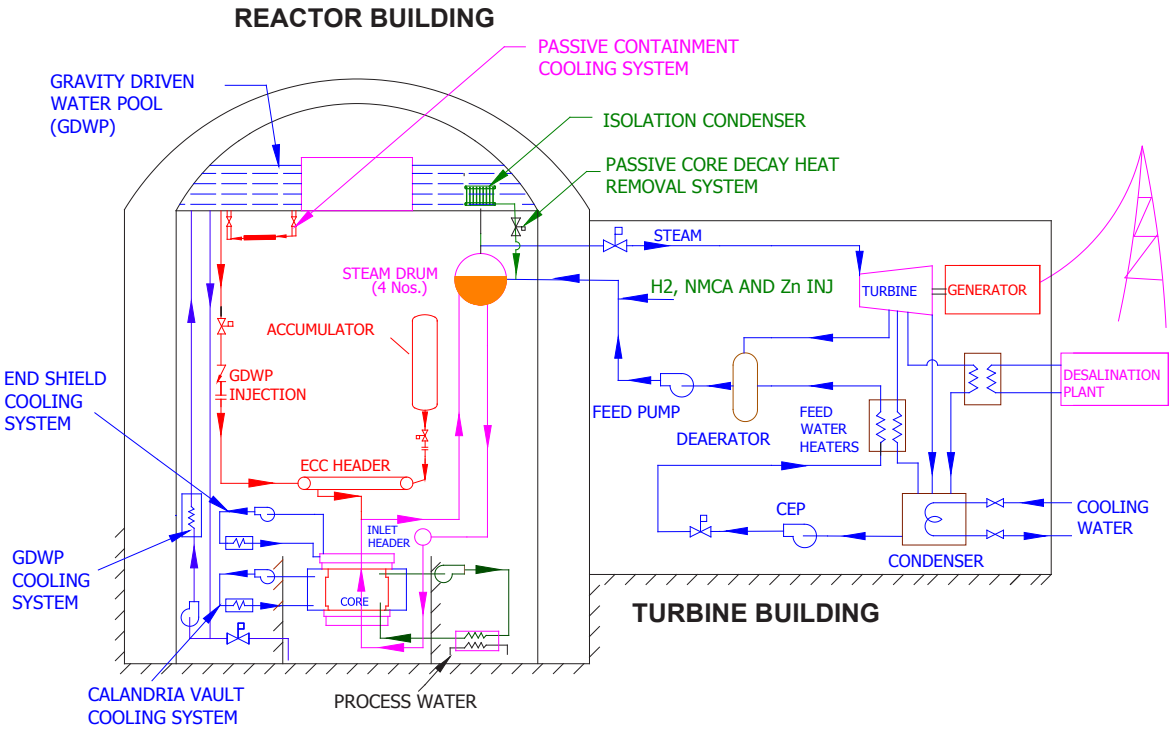


Figure 2: Schematic diagram of AHWR (Reproduced courtesy of BARC)

Reactivity control

The AHWR reactivity control is performed through the use of twenty four (24) numbers of control rods, i.e., eight (8) absorber rod, eight (8) regulating rods and eight (8) shim rods respectively. For the reactor control and shutdown, tantalum alloy based control and shut-off rods are provided. The system consists of a set of shut-off rods, and held on top of the reactor core by individual electro-magnets. These shut-off rods are passively released under abnormal conditions. This is a fail-safe system, so that in case of loss of power from batteries, the shut-off rods would fall under gravity and shutdown the reactor.

Reactor coolant system

The AHWR primary cooling mechanism under normal operating condition and shutdown condition is by natural circulation of coolant. The reactor coolant system (RCS) has been designed to ensure adequate cooling of reactor core under all operational states, and during and following all postulated off normal conditions. The calandria, located below ground level contains vertical coolant channels in which the coolant picks up heat from the fuel assemblies suspended inside the pressure tubes. The naturally circulated coolant flow through the tail pipes to the steam drums, where steam is separated for running the turbine cycle. The four (4) steam drums, receive feed water at stipulated temperature to provide optimum sub-cooling at reactor inlet. Four (4) down-comers from each steam drums bring the flow to a circular inlet header which distributes the flow to each of the 452 coolant channels through individual feeders.

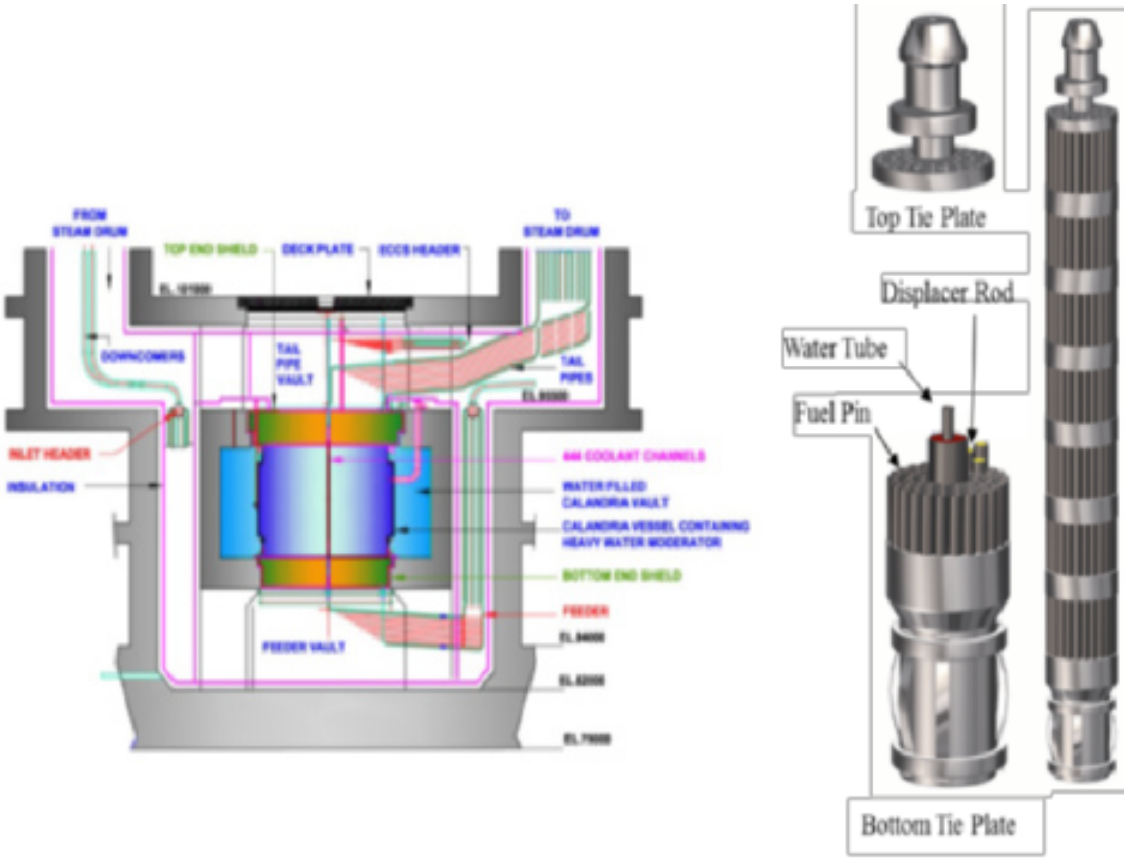


Figure 3: AHWR reactor block and fuel cluster (Reproduced courtesy of BARC)

Steam drum

Steam drum is a horizontally mounted cylindrical vessel closed at both ends by the tori spherical heads. The two-phase steam-water mixture produced in the reactor core enters the steam drums through tail pipes connected to the coolant channels. The feed water enters the steam drum through a sparger, which runs along the length of the steam drum and located in the space between the partition plates. Steam is taken out from each steam drum through outlet nozzles located on the top of the steam drum.

5. Safety Features

Engineered safety system approach and configuration

Several advanced safety features are adopted in the AHWR design to enhance the safety and eliminate any potential impact in the public domain in the case of safety events. The AHWR design also incorporates several passive systems for performing safety-related functions in the event of an accident.

Decay heat removal system

The shutdown cooling system is capable to remove the decay heat and cool down the main heat transport system (MHTS).The main steam condenser (MSC) removes the decay heat and cool down the MHTS. Normal cool down from 558 K to 423 K is achieved by MSC or passively by isolation condensers (ICs), when the MSC is not available. During hot shut down condition, steam generated due to decay heat is fed to the MSC. This steam after condensation is pumped back to the steam drums by feed water pumps. The ICs submerged in the GDWP dissipates the heat to the GDWP.

Emergency core cooling system
The emergency core cooling system (ECCS) consists of four (4) independent circuits and is actuated in passive mode as depressurisation of the MHT system progresses following a LOCA. ECCS operation consists of high pressure injection system using accumulators and a low pressure injection system using GDWP as source of water in a sequential manner. The AHWR design provides a grace period of 7 days for absence of any operator or powered actions in the event of accident.

Gravity driven water pool
The gravity driven water pool (GDWP), a part of the ECCS is a large pool of water containing 8000 m^3 of water inventory, located in the dome region of the reactor building to accommodate the cooling requirements of different systems during various reactor conditions. The GDWP consists of eight (8) interconnected compartments. The GDWP is provided for:

- Removal of core decay heat during shutdown (hot and cold shutdown) by condensation of steam flowing through ICs that are submerged in GDWP;
- Direct injection of low pressure coolant into the core after LOCA for 3 days;
- Vapour suppression during LOCA;
- Removal of containment heat by condensation of steam from the air-steam mixture across the PCC tubes during and after LOCA.
- Removal of heat from moderator and endshield in case of SBO.

The GDWP recirculation and cooling system is provided for cooling the GDWP inventory catering to the above requirements as well as recirculating, filling and draining the water in each compartment of the GDWP. It consists of four (4) heat exchangers, four (4) pumps, filters, ion exchangers and a chemical addition tank to maintain the water chemistry.

Reactor protection system
The reactor protection system comprise of two (2) independent fast acting shutdown systems (SDS).SDS-1 is based on mechanical shut-off rods with boron carbide based absorbers in 33 lattice positions, providing a total negative reactivity worth of 74 mk with all rods inserted, and a worth of 51 mk with maximum two (2) worth rods not available. SDS-2 is based on a liquid poison injection into the moderator. In addition, a pressurized addition of poison passively driven by steam pressure takes place in the event of over pressure in the MHT system. For a long-term sub-criticality control, there is a provision to add boron to the moderator.

Containment system
The containment system houses the reactor, the primary coolant and moderator systems and other systems connected with steam generation. The containment system is designed with passive and inherent safety systems. The primary containment is designed with a diameter of 52 m, height of 75.5 m (54.5 m above ground) and a design pressure/temperature of 0.274 MPa/410 K. Double containment cylindrical concrete structures, roofed with two segmental concrete domes are provided. The GDWP inventory is provided for removal of containment heat by condensation of steam from the air-steam mixture across the PCC tubes, during and after LOCA. Following a large break LOCA the double containment, passive containment isolation system and passive containment cooling system help in passively bringing down the containment pressure and in minimizing any releases from the containment. For the purpose of containment design, a double-ended guillotine rupture of the 600 mm diameter inlet header has been considered as the design basis accident. The hydrogen mitigation system and hard vent for venting of containment and retention of fission products is incorporated in the

primary containment in response to the Fukushima Daiichi event.

6. Plant safety and operational performances

The AHWR is designed to operate efficiently at all operating conditions using natural circulation as the means of providing core coolant flow. The natural circulation flow rate depends on the operating conditions such as the operating pressure, power and feed water flow rate and its temperature. The natural circulation flow rate has also been predicted at partial power conditions considering the feed water temperature variation and its effect. A separate cold start-up procedure for the reactor is required to avoid low power flow oscillations. As per the start-up scheme, boiling in the MHTS is permitted only at 7.0 MPa. The desired pressure in the steam drum is maintained by supplying steam from the start-up boiler. Under normal operating conditions, the pressure in the steam drum is maintained at 7.0 MPa. The feed water with the same mass flow rate enters the steam drum at 403 K. The primary circulation flow rate is of the order of 2145 kg/s maintaining the average steam exit quality at about 19%.The target lifetime load factor and availability factors for the AHWR are 80%and 90%, respectively.

7. Instrumentation and controls 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. The general concepts and design principles followed are redundancy, diversity, fail safe behaviour, fault tolerance, testability and maintainability. The I&C system is designed to regulate the reactor power and process variables of the plant within limits under all operating conditions and postulated emergency situations. The 452 channels in the AHWR are instrumented with in-house developed bi-directional venture flow meters for monitoring the forward flow and identify flow reversal. The design objectives met by the philosophy of defence-in-depth resulted in high reliability and availability of I&C systems, fulfilling stringent safety and operational requirements.

8. Plant Arrangement

Reactor building

The reactor building is a double containment cylindrical concrete structure roofed by two segmental concrete domes. The double containment structure besides containing and controlling release of effluents also provides necessary radiation shielding to the surroundings from the radiation sources present in the reactor building during normal operation. The GDWP is located near the top of the reactor building. The reactor building air supply and exhaust ducts are shaped in the form of U-bends of sufficient height for passive containment isolation.

Control building and structures

The AHWR service building provides nuclear and conventional service facilities like workshops, ventilation intake, compressors and decontamination areas as well as laboratory services of chemical and instrumentation to power station complex. The main control building housing safety group-1 I&C hardware is functionally and physically independent from the backup control building housing safety group-2 I&C hardware.

Balance of plant:

Turbine Generator building

Turbine building accommodates turbine generator, condenser and associated steam cycle equipment required for generation of electricity from steam cycle of the plant. At normal full load, a flow of 408 kg/s, 0.25% wet saturated steam, at a pressure of 6.83 MPa from steam drum is delivered to the turbine throttle. After expansion in the high pressure turbine, steam is exhausted at pressure of 1.025 MPa. Then it passes through an external moisture separator reheater. Subsequently the steam is reheated in bled steam reheaters and enters the low pressure double flow turbine at temperature 503.2K and pressure of 0.956 MPa, where it expands to a condenser back pressure of 8.3kPa.

Electric power systems

AHWR electrical power supply system comprises of off-site supplies and on-site supplies, and associated distribution systems. The onsite power supplies are divided into normal and emergency power supply systems. The emergency power supply systems are only considered as safety related systems. The various unit auxiliary loads connected to the electrical power supply system are broadly divided into two categories: safety and non-safety related loads. Non-safety related loads can endure prolong power failure without affecting the safe operation of plant. The safety related loads depending on their criticality have defined limitation of duration of power failure to ensure safe operation of plant. The emergency power supply system ensures electrical power with specified quality to safety related loads which can take care of safe operation or shut down of plant after a postulated initiating event.

9. Design and Licensing status

The 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. The Nuclear Power Corporation of India Ltd. (NPCIL) has completed a peer review of the design in September 2003. BARC has built several integral and separate effect test facilities for design validation of AHWR. In view of the post Fukushima accident, several features have been incorporated, these include: passive end shield and passive moderator cooling system; hard vent system; passive autocatalytic re-combiners for hydrogen management; and 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. Preliminary safety analysis report (PSAR) is ready for submission to regulatory body after obtaining site clearances from competent authorities.

10. Plant Economics

The low capital cost and advanced safety features and of the AHWR could facilitate its easy deployment. The main design features of the AHWR leading to reduced capital cost per MW(e) are elimination of main coolant pump and associated equipment, substitution of SGs with steam drums, elimination of heavy water recovery and tritium management systems and shop-assembled coolant channel assemblies. The reactor has two important provisions for reducing O&M costs: (a) the elimination of potential for heavy water leakages from the main coolant system such as in conventional PHWRs. This saves the recurring cost for heavy water make up; (b) an easily replaceable coolant channel design.

IRIS (IRIS International Consortium)

1. 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. 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.

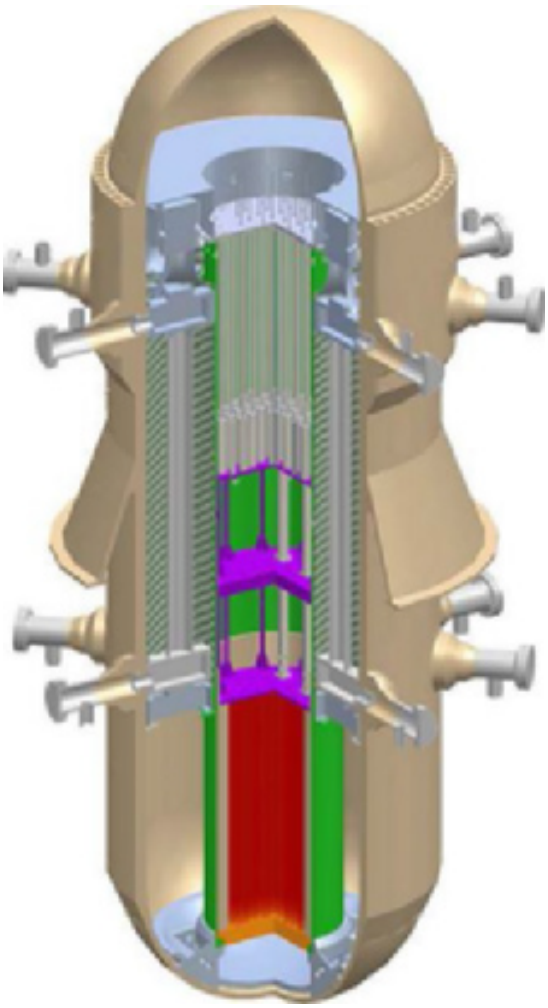


Figure 1: Reactor System Configuration of IRIS
(Reproduced courtesy of POLIMI)

2. Target Application

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.

3. Development Milestones

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

4. General Design Description

Design Philosophy
IRIS is designed to satisfy enhanced safety, improved economics, proliferation resistance and waste minimization.

Nuclear Steam Supply System
All the major NSSS equipment i.e. reactor coolant pumps, steam generators and pressurizer are located inside the RPV, resulting in a more compact configuration and elimination of the large loss-of-coolant accident.

Reactor Core
The IRIS core is an evolutionary design based on conventional UO_2 fuel enriched to 4.95%. An IRIS fuel assembly consists of 264 fuel rods with a 0.374 in. outer diameter in a 17×17 square array. The central position is reserved for in-core instrumentation, and 24 positions have guide thimbles for the control rods. Low-power density is achieved by employing a core configuration consisting of 89 fuel assemblies with a 14-ft (4.267m) active fuel height, and a nominal thermal power of 1000 MW. The core is designed for a 3–3.5-year cycle with half-core reload to optimize the overall fuel economics while maximizing the discharge burnup. In addition, a 4-year straight burn fuel cycle can also be implemented to improve the overall plant availability, but at the expense of a somewhat reduced discharge burnup.

Reactivity Control
Reactivity control in IRIS is achieved through solid burnable absorbers, control rods, and the use of a limited amount of soluble boron in the reactor coolant. The reduced use of soluble boron makes the moderator temperature coefficient more negative, thus increasing inherent safety. Control rod drive mechanisms (CRDMs) are located inside the vessel, in the region above the core and surrounded by the steam generators. Their advantages are in safety and operation. Safety-wise, the uncontrolled rod ejection accident is eliminated because there is no potential 2000-psi differential pressure to drive out the CRDM extension shafts. Operation-wise, the absence of CRDM nozzle penetrations in the upper head eliminates all the operational problems related with corrosion cracking of these nozzle welds and seals.

Reactor Pressure Vessel and Internals
The IRIS reactor vessel houses not only the nuclear fuel and control rods, but also all the major reactor coolant system components: eight small, spool type, reactor coolant pumps; eight modular, helical coil, once through steam generators; a pressurizer located in the RV upper head; the control rod drive mechanisms; and, a steel reflector which surrounds the core and improves neutron economy, as well as it provides additional internal shielding. This integral RV arrangement eliminates the individual component pressure vessels and large connecting loop piping between them, resulting in a more compact configuration and in the elimination of the large loss-of-coolant accident as a design basis event. Because the IRIS integral vessel contains all the RCS components, it is larger than the RV of a traditional loop-type PWR. It has an internal diameter of 6.21 m and an overall height of 22.2 m including the closure head.

Reactor Coolant System

The integral reactor coolant system of IRIS consists of 8 helical-coil steam generators and 8 spool type primary coolant pumps. The motor and pump consist of two concentric cylinders, where the outer ring is the stationary stator and the inner ring is the rotor that carries high specific speed pump impellers. The spool type pump is located entirely within the reactor vessel, with only small penetrations for the electrical power cables and for water cooling supply and return. Water flows upwards through the core and then through the riser region (defined by the extended core barrel). At the top of the riser, the coolant is directed into the upper part of the annular plenum between the extended core barrel and the RV inside wall, where the suction of the reactor coolant pumps is located. The flow from each pump is directed downward through its associated helical coil steam generator module. The primary flow path continues down through the annular downcomer region outside the core to the lower plenum and then back to the core completing the circuit.

Steam Generator

The IRIS has once-through steam generators (OTSGs) with helical-coil tube bundle design with the primary fluid outside the tubes. Eight steam generator modules are located in the annular space between the core barrel and the reactor vessel. Each IRIS SG module consists of a central inner column which supports the tubes, the lower feed water header and the upper steam header. The enveloping outer diameter of the tube bundle is 1.64 m. Each SG has 656 tubes, and the tubes and headers are designed for the full external RCS pressure. The tubes are connected to the vertical sides of the lower feedwater header and the upper steam header. The SG is supported from the RV wall and the headers are bolted to the vessel from the inside of the feed inlet and steam outlet pipes. The 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.

Pressurizer

The IRIS pressurizer is integrated into the upper head of the reactor vessel. The pressurizer region is defined by an insulated, inverted top-hat structure that divides the circulating reactor coolant flow path from the saturated pressurizer water. This structure includes a closed cell insulation to minimize the heat transfer between the hotter pressurizer fluid and the subcooled primary water. Annular heater rods are located in the bottom portion of the inverted top-hat which contains holes to allow water insurge and outsurge to/from the pressurizer region. These surge holes are located just below the heater rods so that insurge fluid flows up along the heater elements. By utilizing the upper head region of the reactor vessel, the IRIS pressurizer provides a very large water and steam volume, as compared to plants with a traditional, separate, pressurizer vessel. The IRIS pressurizer has a total volume of ~71 m^3, which includes a steam volume of 49 m^3.

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)	14000 (four units layout)
Coolant/moderator	Light water
Primary circulation	Forced circulation
System pressure (MPa)	15.5
Core inlet/exit temperatures (^oC)	292/ 330
Main Reactivity Control Mechanism	ICRDM (Internally driven Control Rods)
RPV Height (m)	21.3
RPV Diameter (m)	6.2
Module Weight (metric ton)	–
Configuration of Reactor Coolant System	Integrated
Power Conversion Process	Indirect Rankine cycle
Fuel Type/Assembly Array	UO_2/MOX/17x17 square
Fuel Assembly 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)
Cogeneration Capability	Possible
Approach to Engineered Safety Systems	Passive
Number of safety trains	4
Refueling Outage (days)	-
Distinguishing features	Integral primary system configuration
Modules per Plant	1 to 4
Target Construction Duration (months)	36
Seismic design (g)	0.30
Predicted core damage frequency (per reactor year)	1E–8
Design Status	Basic design

5. 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. This provides a high level of defence in depth that may allow IRIS to claim no need for an emergency response zone. The auxiliary building of IRIS is 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.

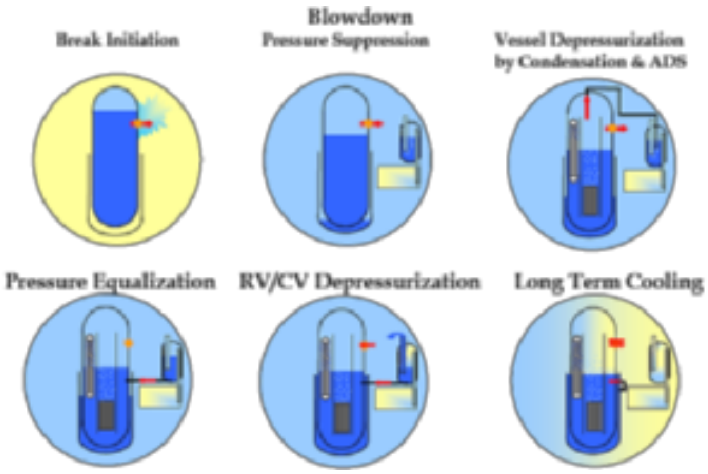


Figure 2: SBLOCA safety strategy (Reproduced courtesy of POLIMI)

Engineered Safety System Approach and Configuration

IRIS has passive emergency heat removal system (EHRS) made of four independent subsystems, each of which has a horizontal, U-tube heat exchanger connected to a separate SG feed/steam line. These heat exchangers are immersed in the refueling water storage tank (RWST) located outside the containment structure. The RWST water provides the heat sink to the environment for the EHRS heat exchangers. The EHRS is sized so that a single subsystem can provide core decay heat removal in the case of a loss of secondary system heat removal capability. The EHRS operates in natural circulation, removing heat from the primary system through the steam generators heat transfer surface, condensing the steam produced in the EHRS heat exchanger, transferring the heat to the RWST water, and returning the condensate back to the SG. The EHRS provides both the main post-LOCA depressurization (depressurization without loss of mass) of the primary system and the core cooling functions. It performs these functions by condensing the steam produced by the core directly inside the reactor vessel. This minimizes the break flow and actually reverses it for a portion of the LOCA response, while transferring the decay heat to the environment. The safety strategy of IRIS provides a diverse means of core shutdown by makeup of borated water from the emergency boration tanks (EBT) in addition to the control rods; also, the EHRS provides a means of core cooling and heat removal to the environment in the event that normally

available active systems are not available. In the event of a significant loss of primary-side water inventory, the primary line of defence for IRIS is represented by the large coolant inventory in the reactor vessel and the fact that EHRS operation limits the loss of mass, thus maintaining a sufficient inventory in the primary system and guaranteeing that the core will remain covered for all postulated events

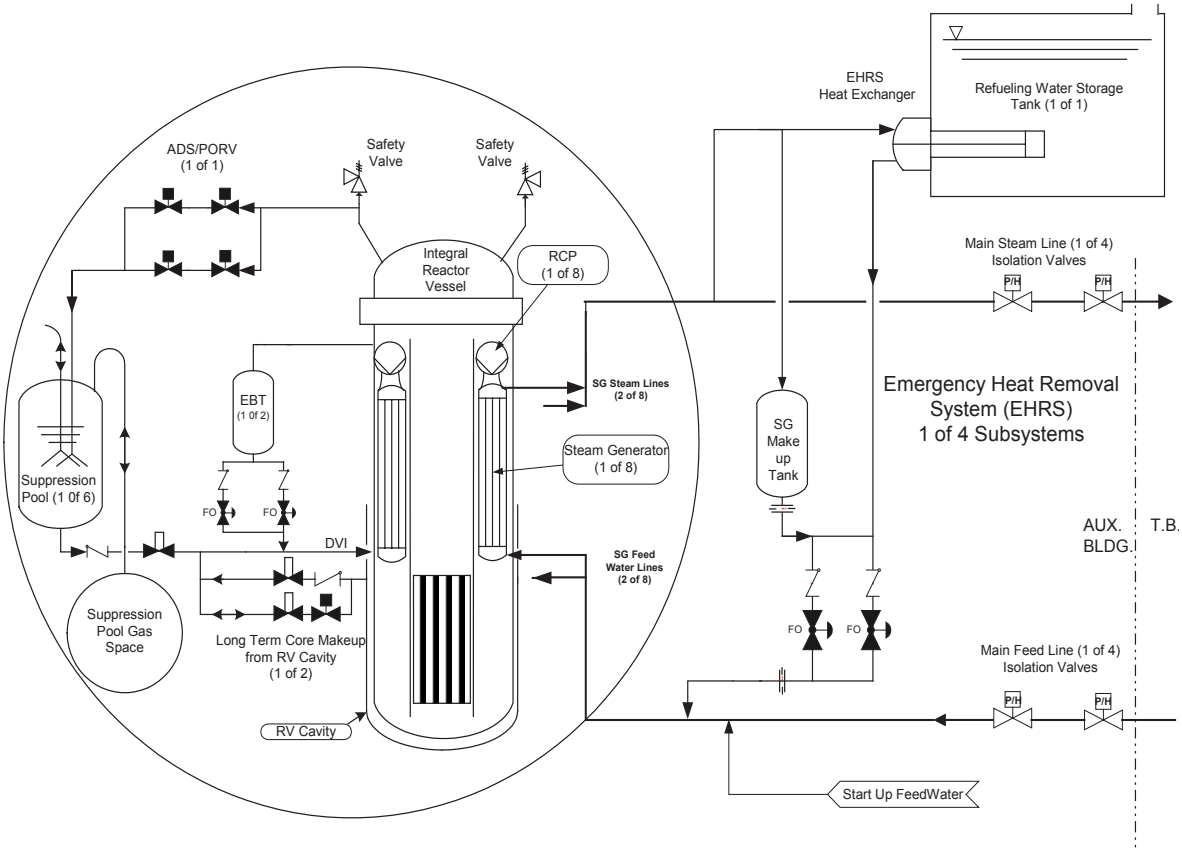


Figure 3: Engineered safety features of IRIS (Reproduced courtesy of POLIMI)

Containment system

Because the IRIS integral RV configuration eliminates the loop piping and the externally located steam generators, pumps and pressurizer with their individual vessels, the footprint of the patent-pending IRIS containment system is greatly reduced. This size reduction, combined with the spherical geometry, results in a design pressure capability at least three times higher than a typical loop reactor cylindrical containment, assuming the same metal thickness and stress level in the shell. The current layout features a spherical, steel containment vessel (CV) that is 25 m (82 ft.) in diameter as shown in *Figure. 4*. The CV is constructed of 1-¾ in. steel plate and has a design pressure capability of 1.4 MPa (~190 psig). The CV has a bolted and flanged closure head at the top that provides access to the RV upper head flange and bolting. Refueling of the reactor is accomplished by removing the containment vessel closure head, installing a sealing collar between the CV and RV, and removing the RV head. The refueling cavity above the containment and RV is then flooded, and the RV internals are removed and stored in the refueling cavity. Fuel assemblies are vertically lifted from the RV directly into a fuel handling and storage area, using a refueling machine located directly above the CV. Thus, no refueling equipment is required inside containment and the single refueling machine is used for all fuel movement activities. *Figure 4* also shows the pressure suppression pool that limits the containment peak pressure to well

below the CV design pressure. The suppression pool water is elevated such that it provides a potential source of elevated gravity driven makeup water to the RV. Also shown is the RV flood-up cavity formed by the containment internal structure. The flood-up level is 9 m and ensures that the lower section of the RV, where the core is located, is surrounded by water following any postulated accident. The water flood-up height is sufficient to provide long-term gravity makeup, so that the RV water inventory is maintained above the core for an indefinite period of time. It also provides sufficient heat removal from the external RV surface to prevent any vessel failure following beyond design basis scenarios.

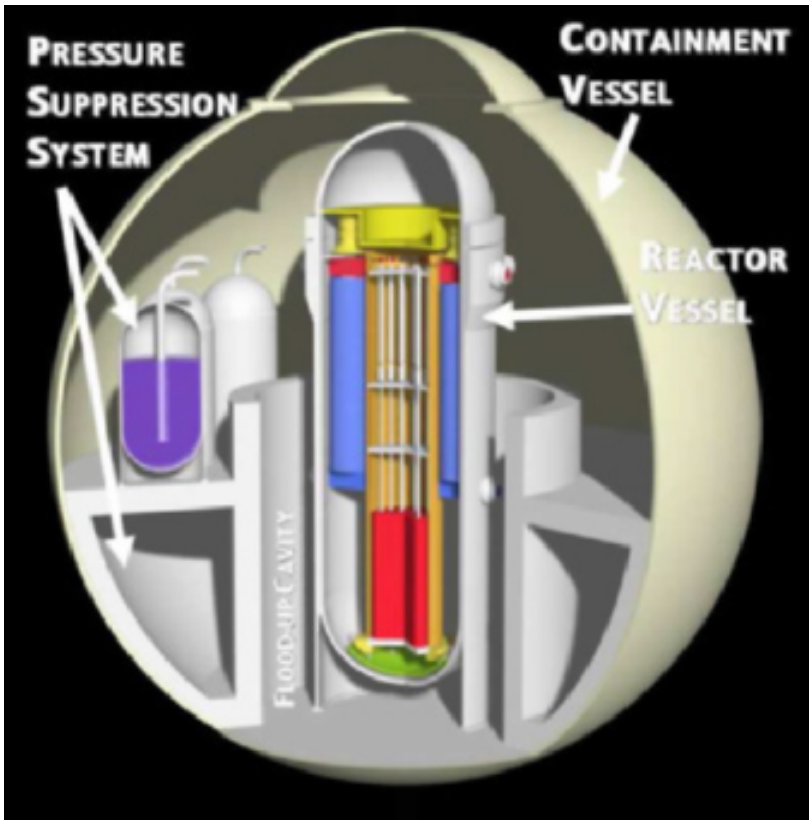


Figure 4: IRIS spherical steel containment arrangement.
(Reproduced courtesy of POLIMI)

6. Plant safety and Operational Performances

The IRIS design provides for multiple levels of defense for accident mitigation, resulting in extremely low core damage probabilities. In addition to the traditional DID levels (barriers, redundancy, diversity, etc.) IRIS introduces a very basic level of DID, i.e., elimination by design of accident initiators or reduction of their consequences/probability. Furthermore, a distinguishing characteristic of IRIS is its capability of operating with long cycles. Even though the reference design features a two-batch, 3-year fuel cycle, selected on the basis of ease of licensing and US utilities preference, IRIS is capable of eventually operating in straight burn with a core lifetime of up to 8 years. However, the significant advantages connected with a long refueling period in reducing O&M costs are lost if the reactor still has to be shut down on a 18–24-month interval for routine maintenance and inspection. Thus, first and foremost, the IRIS primary system components are designed to have very high reliability to decrease the incidence of equipment failures and reduce the frequency of required inspections or repairs. Next, IRIS has been designed to extend the need for scheduled maintenance outages to at least 48 months. Because of the 4-year maintenance

cycle capability, the capacity factor of IRIS is expected to comfortably satisfy and exceed the 95% target, and personnel requirements are expected to be significantly reduced. Both considerations will result in decreased O&M costs.

7. Plant Arrangement

Almost half of the IRIS containment vessel is located below ground, thus leaving only about 15 m above the ground (i.e., several times less than the containment of a large LWR). This very low profile makes IRIS an extremely difficult target for aircraft flying terrorists; in addition, the IRIS containment is inconspicuously housed in and protected by the reactor building. The cost of putting the entire reactor underground was evaluated; it was judged to be prohibitive for a competitive entry to the power market and unnecessary since the IRIS design characteristics are such to offer both an economic and very effective approach to this problem.

8. Design and Licensing 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)

1. Introduction

DMS stands for Double MS: Modular Simplified and Medium Small Reactor. The design concept is developed by Hitachi-GE Nuclear Energy under the sponsorship of the Japan Atomic Power Company (JAPC). The DMS is a small-sized boiling water reactor (BWR) that generates a capacity of 840 MW(th) or about 300 MW(e). The DMS reactor aims to optimize the design according to the power output and achieve high economy by utilizing proven technologies of existing reactors. The heat produced in the core is removed by natural circulation of the coolant, thus eliminating the recirculation pumps and their driving power sources. 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. 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 factory-fabricated module reduces the construction period and also enables the modules to be transported to the site.

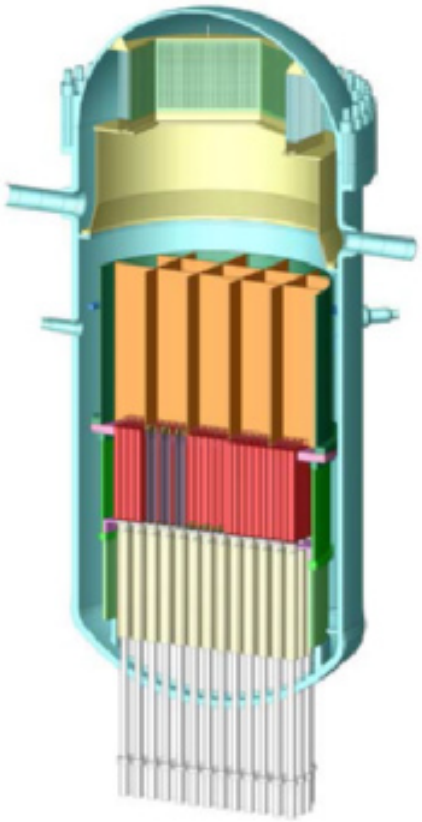


Figure 1: Reactor System Configuration of DMS
(Reproduced courtesy of Hitachi-GE Nuclear Energy)

2. Target Application

A small-to-medium sized BWR is suitable for remote regions with less developed grids and infrastructures. DMS design provides a nonelectric use of energy such as for district heating, mining (oil sand extraction/steam assisted gravity drainage) and desalination.

3. Development Milestones

2000-2004	Conceptual design
2014	Basic design (pre-licensing)
2017~	Design Review or Design Certification
2020~	Proposal to Customer or Commercial Bid
2030~	Commercial Operation

4. General Design Description

The main features of the DMS reactor include:

- Simplified reactor core structures: Application of small sized natural circulation reactor, the elimination of steam separator and the application of small sized RPV.
- Simplified systems and safety systems: Natural circulation eliminates the reactor internal pumps and proper equipment configuration and simplification is achieved by integrating systems/equipment. e.g., two main steam lines, the application of passive system, common use of systems, simplified ECCS structure, etc. Active core injection system including emergency generator is simplified by taking advantage of the large coolant inventory in the RPV.
- Rationalized layout and construction method: By applying simplified systems and compact primary containment vessel, the volume of building is reduced to a large extent. The construction period is also reduced by applying large-scale modular construction in which equipment, facilities and building structure are integrated.

Design Philosophy

The DMS is developed with the concept of high-economy small sized reactor of short construction period to meet the diversified market needs. The design is based on a small sized reactor pressure vessel, simplified safety systems, rationalized layout and architectural design. The reactor is based on a proven technology with experience from existing BWRs, intended for using systems and equipment that requires no large-scale development. The DMS is designed to obtain a high safety level equivalent to the existing reactors with optimized operations and maintenance performance in accordance with the level of output power.

Nuclear Steam Supply System

The nuclear steam supply system (NSSS) of the DMS is a direct cycle, where steam, generated in the core, goes into a turbine directly. NSSS includes the reactor core and internals, reactor pressure vessel and the coolant/steam piping within the containment system. Primary pump, as well as primary re-circulation loop, is not equipped, due to the feature of natural circulation system. These system features contribute to remove a potential of LOCA. Also, all of the large sized piping connected to the RPV is designed to be located above a top of an active fuel so that the DMS core is able to be covered by coolant even if the most severe loss of coolant accident (LOCA) occurs. The NSSS of the DMS is designed to be able to remove core power by natural circulation core flow during normal operation, and the core can be also cooled by natural circulation flow even if an anticipated transient or accident occurs.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	Hitachi-GE Nuclear Energy
Country of origin	Japan
Reactor type	Boiling Water Reactor
Electrical capacity (MW(e))	About 300
Thermal capacity (MW(th))	840
Expected capacity factor	> 87
Design life (years)	60
Plant footprint (m^2)	--
Coolant/moderator	Light water
Primary circulation	Natural circulation
System pressure (MPa)	7.17
Feedwater /Core exit temperatures (^oC)	186 / 287
Main reactivity control mechanism	Control rod drive, motor operated in normal operation.
RPV height (m)	15
RPV diameter (m)	4.8
Module weight (metric ton)	500~600 ton
Configuration of reactor coolant system	Integral type
Power conversion process	Direct Rankine cycle
Fuel type/assembly array	UO_2 pellet/10x10 square configuration with channel box
Fuel assembly active length (m)	2.0
Number of fuel assemblies	400 (Short length fuel assembly)
Fuel enrichment (%)	<5
Fuel burnup (GWd/ton)	<60
Fuel cycle (months)	24
Cogeneration capability	Yes
Approach to engineered safety systems	Hybrid (Passive + Active) system
Number of safety trains	3 active and 1 passive system
Refuelling outage (days)	30 (target)
Distinguishing features	Simple reactor design, natural circulation system, hybrid safety system, multipurpose use of energy.
Modules per plant	1
Target construction duration (months)	~24
Seismic design	0.45g
Core damage frequency (per reactor-year)	5.0E–8
Design Status	Basic design

Reactor Core

The reactor core is loaded with 400 fuel bundles. The fuel active length is 2.0 m with the enrichment of less than 5 wt%. The short active fuel length reduces the pressure drop in core and enables natural circulation. The short fuel length increases the number of fuel assemblies necessary to secure the required thermal output increases, which results in increasing the diameter of RPV and the number of control rod drives, but the flow rate of natural circulation

can be reduced, making possible the reduction of RPV height. The core power density is about 44 MW/m^3. The core can produce energy with the refuelling period of 24 months.

Reactivity Control

The DMS has two kinds of diverse rector shutdown systems (i.e., control rod (CR)/control rod driving system (CRD) and standby liquid control system (SLCS)). CR uses B_4C or Hf as a neutron absorber and it is designed to be inserted from the bottom of the core. Every CR has an independent CRD per each at the bottom of the reactor vessel. Since the CRD of DMS has a motor-controlled fine motion capability, it is also called Fine motion CRD (FMCRD) and it controls a positioning of CR, including insertion and withdrawal. DMS has 97 CRs and FMCRDs. FMCRD has two kind of independent operation mode, one is fine motion control by the electric motor, and the other is rapid motion (scram) by the hydraulic pressure. In the normal operation, the reactivity in the core is controlled by the fine motion feature of FMCRD.

Reactor Pressure Vessel and Internals

The size of the RPV is one of the main factors that determine the size of primary containment vessel (PCV) and also greatly influence construction cost of the reactor building. The small sized RPV in the DMS reactor is achieved by simplifying core internals through small sized reactor core and natural circulation, eliminating steam separator. The flat core concept with short-length and large core diameter is adopted to the DMS and the power density of the DMS core is reduced to about 44MW/m^3.The low power density results in a moderate evaporation rate and lower steam velocity in the upper plenum of the 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.

Reactor Coolant System

The DMS reactor primary cooling mechanism under normal operating condition and shutdown condition is by natural circulation of coolant. In order to enhance natural circulation driving force, divided chimney of about 3m height is adopted above the core. The reactor coolant system (RCS) is designed to ensure adequate cooling of reactor core under all operational states, and during and following all postulated off normal conditions. A large coolant inventory is achieved in the reactor due to the increased height and diameter of the RPV. Like the conventional 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.

5. Safety Features

A large coolant inventory in achieved in the DMS compared with the forced circulation type reactor of the same output level, because the height of RPV is increased to secure the driving power of natural circulation and the diameter of RPV is increased due to increased number of fuel assemblies. These results in relieving the influence on reactor core in case of transient phenomena or medium and small piping break accident, so that high pressure core flooder (HPCF) system, which is equipped in a conventional BWR, can be eliminated. At the same time, because the load of emergency generator is mostly shared by the motor capacity of conventional HPCF, it becomes possible not only to eliminate HPCF system pumps but also to reduce the capacity of emergency generator. By optimizing the configuration of systems in this way, it becomes possible to simplify facilities to a large extent while keeping safety at the level equivalent to conventional reactors.

Engineered Safety System Approach and Configuration

As a defence-in-depth measure, enhanced hybrid safety systems that combine passive and active methods are adopted. The safety system configuration of DMS has been rationally simplified compared to a conventional large BWR (ABWR). There are four main distinctive features: (1) High pressure core flooder (HPCF) equipped in conventional ABWR eliminated due to the larger coolant inventory in the RPV of the DMS than ABWR; (2) Isolation condenser (IC) and passive containment cooling system (PCCS) were 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. Though required time for start-up of GTG is longer than that of D/G, 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°C even against the most severe DBA. It should be noted that optimization of the safety system configuration continues under the discussion with potential customers and safety regulators to meet their expectations

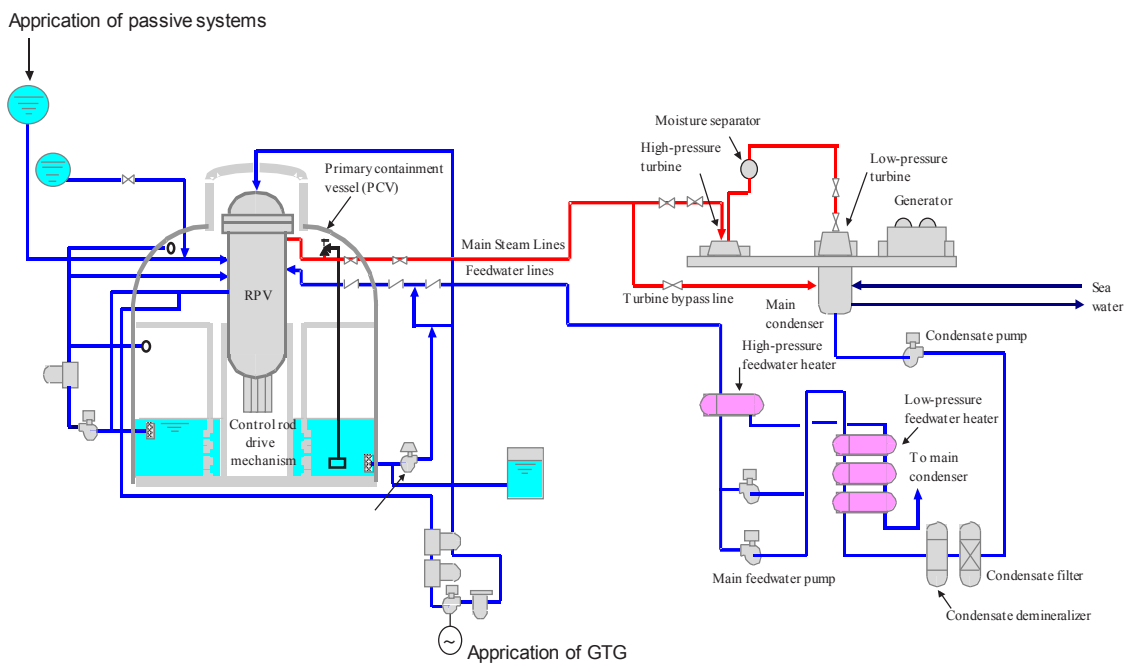


Figure 2: Main configurations and safety features of DMS
(Reproduced courtesy of Hitachi-GE Nuclear Energy)

Decay Heat Removal System

The residual heat removal system (RHR) includes a number of pumps and heat exchangers which can be used to cool water in the RPV or the suppression pool (S/P) in the PCV. The RHR can remove residual heat not only during normal shutdown and reactor servicing period but also during an accident such as LOCA. The PCV cooling is accomplished by extracting

and cooling the S/P water, and injecting cooled water back to either the S/P injection line or the containment spray lines. The IC and PCCS are equipped in the DMS as passive safety systems without using any AC power. The IC and PCCS can condense steam generated in the RPV and steam released from the RPV to the PCV via ruptured piping during LOCA, respectively. The heat exchangers of the IC and PCCS can be cooled by the dedicated water pool located above the PCV, which is filled with an amount of water enough to remove decay heat at least 10 days passively.

Containment System

Steel containment is applied to achieve the design pressure (427 kPa[g]) equivalent to that of Mark-I type containment, and the quantity of material is reduced by reducing the diameter and height of PCV by adopting dish shape drywell and eccentric RPV arrangement. The concept of compact PCV that improves constructability is established. The pressure suppression containment that is experienced in BWR and ABWR was applied while compactness was aimed at by eliminating steam separator, thus reducing the height of RPV and the number of main steam pipes. The decrease in PCV height is achieved by reducing the active fuel length of the DMS core, which is about 2 meters compared with 3.7 meters in the conventional BWR. The PCV is inserted by nitrogen during normal operation, therefore, hydrogen combustion in the PCV in early timing is practically eliminated. For a long-term accident, a few number of passive autocatalytic re-combiners (PARs) are planned to equip in the PCV to react hydrogen and oxygen generated due to water radiolysis.

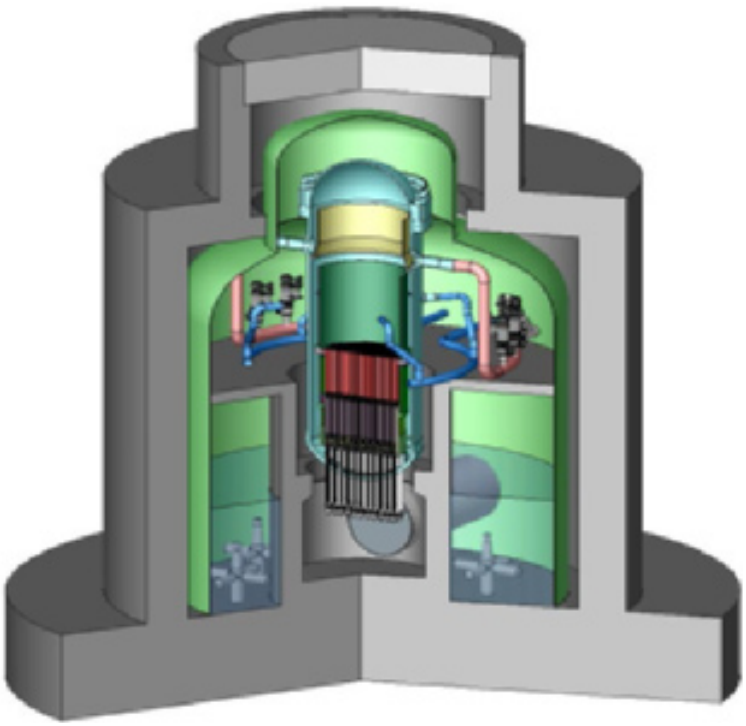


Figure 3: Compact primary containment vessel of DMS
(Reproduced courtesy of Hitachi-GE Nuclear Energy)

6. Plant safety and Operational Performances

The performance of the plant is improved by applying the main steam isolation valve of low pressure loss developed for large sized reactors, for which the number of main steam pipes are reduced to two in order to make the bore diameter of the pipes equivalent to that of conventional BWR, as well as minimizing PCV by relieving the restrictions on the layout in

primary containment vessel. In addition, considering the trade-offs of thermal efficiency and installation cost, facilities are optimized to realize the concept of simplifying facilities to a large extent.

7. Instrumentation and Control systems

The digital I&C systems (e.g. microprocessors, field programmable gate arrays (FPGAs)) are mainly adopted in the DMS design and make use of fault detection and fault tolerance. Diversity is also important particularly in providing a countermeasure against common cause failure (CCF). Hardwired back-up safety system based on analogue technology is planned to be installed to the DMS for diversity to mitigate influence of CCF of the digital I&C system. The I&C system includes the safety system logic and control (SSLC), the plant control systems, the hardwired back-up safety system, the auxiliary control system, and the plant computer system. The Reactor Protection system (RPS) which initiates rapid and automatic shutdown (scram), Emergency Core Cooling system / Emergency Safety features (ECCS/ESF) which initiates ECCS are included in the SSLC.

8. Plant Arrangement

The concept of building layout and plant construction method is established, aiming at reducing the quantity of construction material, shortening the construction period and standardizing the plant design. The reduction in construction cost is established by applying steel plate reinforced concrete structure, simplifying the shape of building and the building structure.

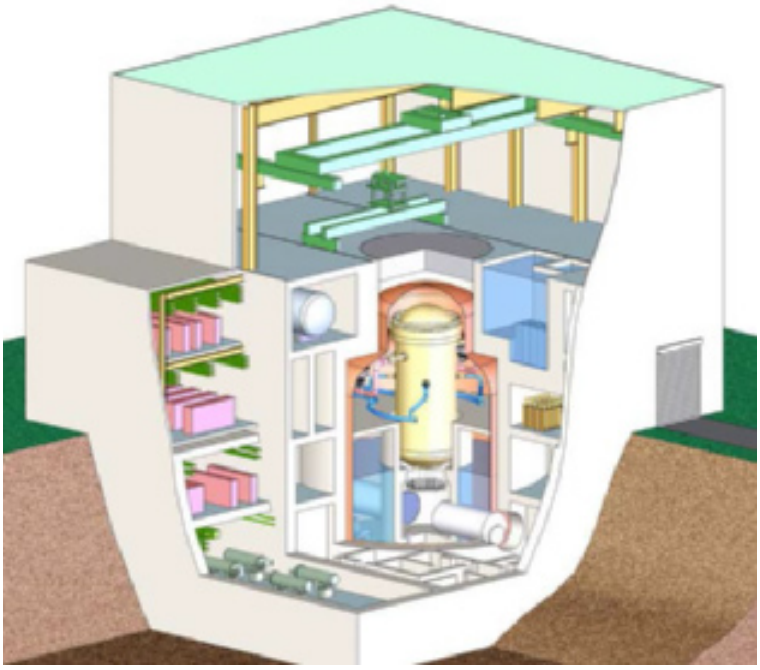


Figure 4: Standardized reactor building
(Reproduced courtesy of Hitachi-GE Nuclear Energy)

Reactor building

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 centre but at an eccentric centre 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.

Control building

The Control Building (C/B) includes the main control room (MCR), the computer facility, the cable tunnels, some of the plant switchgear, emergency batteries, reactor building water system and a heating ventilating and air conditioning system (HVAC). During the plant normal/emergency operation, the HVAC provides ventilation and air conditioning function. For instance, in case of fire, smoke is discharged to outdoor and fresh air is supplied to the C/B.

Balance of plant:

The advanced BOP system allows the utilization of produced heat for non-electrical applications such as process heat, mining (oil sand extraction) and desalination.

Turbine Generator building

The structure of turbine systems are simplified to a large extent by applying the single casing design that uses 41 inch turbine, which is proper for the output level as well as by using single shell condenser and single train (4-stage heating) feed-water heater. The number of equipment is reduced by integrating high and low pressure condensate pumps and optimizing the configuration of systems.

Electric power systems

The DMS is planned to connect to the external grid via a main connection and a standby connection. The main connection is the connection between the generator transformer (GT) and the external grid. The standby connection is the connection between the auxiliary standby transformer (AST) and the external grid.

9. Design and Licensing Status

At the moment, there is 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 DC in USA, GDA in UK or VDR in Canada. Vendor's design is reviewed in the license process of the Establishment Permit (Construction license) of the NPP construction project).

10. Plant Economics

The total cost per capacity ($/kW) of the DMS is estimated to be equivalent to that of a conventional ABWR.

IMR (Mitsubishi Heavy Industries, Japan)

1. Introduction

The Integrated Modular Water Reactor (IMR) is a medium sized power reactor with a reference output of 1000 MW(th) producing an electrical output of 350 MW(e). The IMR is developed for potential deployment after 2025. IMR employs the hybrid heat transport system (HHTS), a natural circulation system for primary heat transportation. The in-vessel control rod drive mechanism (CRDM) is adopted as the primary means of reactivity control. These design features allow the elimination of the emergency core cooling system (ECCS).

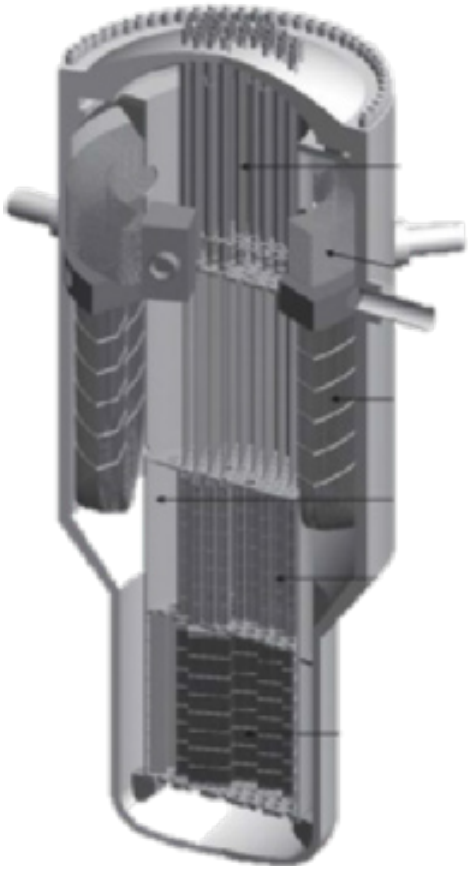


Figure 1: Reactor System Configuration of IMR
(Reproduced courtesy of Mitsubishi Heavy Industries, Ltd.)

2. Target Application

The IMR is primarily designed as a land-based power station module to generate electricity. The capacity of the power station can easily be uprated and adjusted to the demand by installing 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 capacity of grids are small. IMR also has the capability for district heating, seawater desalination, process steam production and so forth. IMR bases on the equipment and systems that require no large-scale development and therefore can rapidly be put to practical use in the market, including the period necessary to obtain authority's approval for the construction and operation of power plant.

3. Development Milestones

1999	IMR started its conceptual design study at Mitsubishi Heavy Industries (MHI).
2001 – 2004	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 experiments.
2005 – 2007	In the second project, the thermal-hydraulic data under natural circulation conditions for the HHTS design were obtained by four series of simulation tests using alternate fluids.
2010 – 2013	Startup transient tests to study the startup flow instability

4. General Design Description

Design Philosophy

The IMR is an integral type water reactor where the primary system components are all installed inside the reactor pressure vessel (RPV). Its features are as follows:

- Main coolant piping and primary coolant pumps are eliminated by adopting integrated natural circulation system;
- Pressurizer is eliminated by adopting the self-pressurization system;
- The CRDMs are located inside the RPV;
- There are two types of steam generators (SGs) inside the RPV. One is located in the vapour portion in the RPV (called SG in vapour (SGV)), and the other is located in the liquid portion in the RPV (called SG in liquid (SGL));
- SGs are also used as decay heat removal heat exchangers during accidents and normal startup and shutdown operations;
- A passive safety system, which does not require any external support (Stand-alone direct heat removal system: SDHS) is adopted;
- ECCSs are not required.

The IMR is a PWR with moderation ratios similar to those of operating PWRs, and thus, its properties of fresh and spent fuel are also similar. This allows for the basic adoption of conventional safeguards procedures and PWR management practices for new and spent fuel. 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.

Nuclear Steam Supply System

The HHTS is employed to transport the fission energy released in the fuel to the SGs by both vapour formation and liquid temperature rise. The energy transported by vapour produces secondary steam in SGV, and the energy transported by liquid temperature rise produces secondary steam in SGL. The SGV also has a function of primary system pressure control, and the SGL has the function of core power control through the core inlet temperature by controlling the feedwater flow rate.

Reactor Core

The IMR has a core consisting of 97 fuel assemblies of 21×21 array with an average enrichment of 4.95% and produces a thermal output of approximately 1000 MW(th). 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.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	Mitsubishi Heavy Industries, Ltd.
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)	-
Coolant/moderator	Light water
Primary circulation	Natural circulation
System pressure (MPa)	15.51
Core inlet/exit temperatures (^oC)	329 / 345
Main reactivity control mechanism	Control rod drive mechanism
RPV height (m)	17
RPV diameter (m)	6
RPV or module weight (metric ton)	-
Configuration of reactor coolant system	Integral type
Power conversion process	Indirect Rankine cycle
Fuel type/assembly array	UO_2 pellet / 21 x 21 square
Fuel assembly active length (m)	2.4
Number of fuel assemblies	97
Fuel enrichment (%)	4.8
Fuel burnup (GWd/ton)	> 40
Fuel cycle (months)	26
Cogeneration capability	Yes
Approach to engineered safety systems	Hybrid (Passive + Active) system
Number of safety trains	4
Refuelling outage (days)	< 30
Distinguishing features	Integral PWR with natural circulation; employs two types in-vessel circular steam generators, one in liquid region, one in vapour region.
Modules per plant	1
Target construction duration (months)	< 42
Seismic design (g)	0.3
Core damage frequency (per reactor-year)	2.90E–7
Design Status	Conceptual design completed.

The cladding material employs Zr–Nb alloy to obtain integrity at a temperature of 345°C and over the long reactor lifetime. To maintain the core thermal margin and to achieve a long fuel cycle, the core power density is reduced to 1/3 - 1/2 of a conventional PWR. The design-refuelling interval is 3 years in 3 batches of fuel replacement. The fuel rod design is the same as that for a conventional PWR.

Reactivity Control
The chemical shim reactivity control is not adopted but both control rods which contain enriched boron 10 and burnable absorbers control the whole reactivity. Control rods whose neutron absorber is 90wt% enriched B_4C perform the reactivity control, and a soluble acid boron system is used for the backup reactor shutdown to avoid corrosion of structural materials by boric acid. The hydrogen to uranium ratio (H:U) is set to five, which is larger than in conventional PWRs, to reduce the pressure drop in the primary circuit. The coolant boils in the upper part of the core and the core outlet void fraction is less than 20% locally and less than 40% in the core to keep bubbly flow conditions. To reduce axial power peaking caused by coolant boiling, the fuel consists of two parts: the upper part with higher enrichment and the lower part with lower enrichment. Additionally, hollow annular pellets are used in the upper part fuel to reduce axial differences of burnup rate. Two types of in-vessel CRDMs are adopted. One is motor driven CRDM, which is applied to the control bank. The basic development of this CRDM has been conducted by the Japan Atomic Power Research Institute (JAERI). This CRDM has the function of controlling reactivity during operation by finely stepping the control rod position. The other is the hydraulic type CRDM. This CRDM has the scram function and applied to the shut-down bank. The control rods connected to this CRDM are moved by hydraulic force from the bottom position to the top, and then held by electro-magnetic force. When the scram signal is initiated, the control rods are released and inserted into the core by gravity by turning off the power to the CRDM.

Reactor Pressure Vessel and Internals
The upper part of the RPV inside diameter is about 6 m in order to accommodate the in-vessel SGs. The inside diameter of the lower part of the RPV is reduced to about 4 m in order to reduce the cold-side water volume. In order to eliminate the necessity for the consideration of LOCA, the largest diameter nozzle connected to the RPV is reduced to less than 10 mm. In addition, the lowest location of the nozzle is above the core to improve the reliability of the RPV. The core is located in the bottom of the RPV and the SGs are located in the upper part of the RPV. Control rod guide assemblies are located above the core and a riser is set above the control rod guide assemblies to enhance the natural circulation.

Reactor Coolant System
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 RPV. 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 RPV 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.

Steam Generators
There are two types of SGs adopted in the IMR. One is the SG in vapour (SGV), which is located above the water level in the RPV. The energy transported by vapour formation generates secondary steam through SGV. In addition, since the vapour in the RPV is condensed by SGV, controlling the feedwater flow rate to SGV can control the RPV pressure. The other is the steam generator in liquid (SGL), which is located in the water in the RPV. The energy transported by liquid temperature rise generates secondary steam through SGL, and also, since the core inlet temperature can be controlled by the amount of heat removal through SGL, the core power can be controlled by controlling the feedwater flow rate to

SGL. By this method, the movement of the control rods for controlling reactor power will be minimized. In *Figure 2* for SGL, a U-type tube bundle is adopted, since it is necessary to minimize pressure drops on both the primary and secondary sides to maintain good natural circulation performance. A C-type steam generator is adopted for SGV, because of the good space utilization in the vapour part of the RPV.

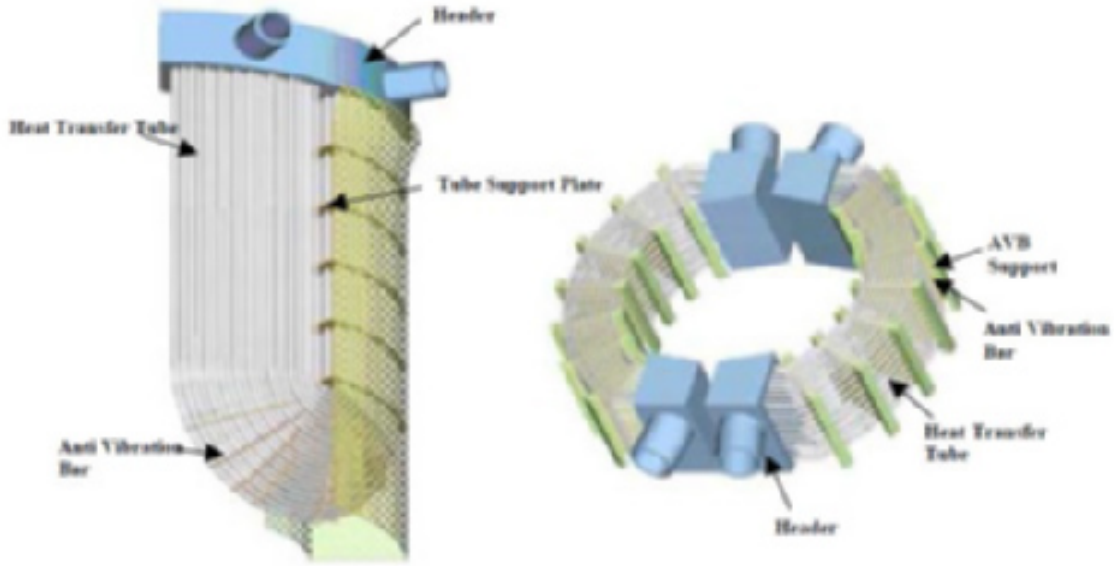


Figure 2: Left: Steam Generator in Liquid Portion; Right: Steam Generator in Vapour Portion (Reproduced courtesy of Mitsubishi Heavy Industries, Ltd.)

Pressurizer
The pressurizer is eliminated by adopting the self-pressurization system.

5. Safety Features

Engineered Safety System Approach and Configuration
Significant improvements have been achieved in the safety goals and safety philosophy of the IMR. By adopting an integral type primary system, accidents that may cause fuel failure, such as loss of coolant accidents (LOCA), rod ejection (R/E), loss-of-flow (LOF) and locked rotor (L/R), are eliminated. Since the diameter of the pipes connected to the primary system (reactor vessel) is limited to less than 10 mm, the water level in the RPV can be maintained at normal levels by water injection from the charging pumps. There are two trains of the SDHS. Therefore, if a malfunction such as SG tube leakage were to occur, system functions would be maintained. The capacity of chemical and volume control system (CVCS) is covered with eight 3/4-inch pipes which are connected to reactor vessel because of the small reactor coolant inventory. No ECCS and containment cooling/spray system is installed. Safety injection systems are eliminated by adopting the SDHS and by limiting the nozzle diameter connected to the primary system. Containment spray system is eliminated by adopting the SDHS. The auxiliary feedwater system is used for startup and shutdown procedures during normal operation. It is also used at the initial stage of the transients caused by the secondary systems. When the plant safety is threatened only by the auxiliary feedwater system, the safety system (SDHS) is actuated. The auxiliary feedwater system is not a safety system, because the plant safety is guaranteed by SDHS. The IMR adopts simplified 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. A simplified CVCS and waste disposal system (WDS) is achieved by the boric-acid free design.

Decay Heat Removal System

For the mitigation system, the SDHS is activated. *Figure 3* shows the concept of SDHS. In this system, the decay heat is removed directly from inside the RPV to the atmosphere. Therefore, even if water leakage occurred and the charging pumps fails to operate, water leakage would be terminated automatically when the pressures inside and outside the RPV are equalized. In the passive steam generator cooler (PSGC), decay heat is removed by water-cooling in the early stage of the accident and then, the heat transfer mode is gradually replaced by air-cooling. Therefore, external support such as water, power and operators are not necessary for maintaining plant safety.

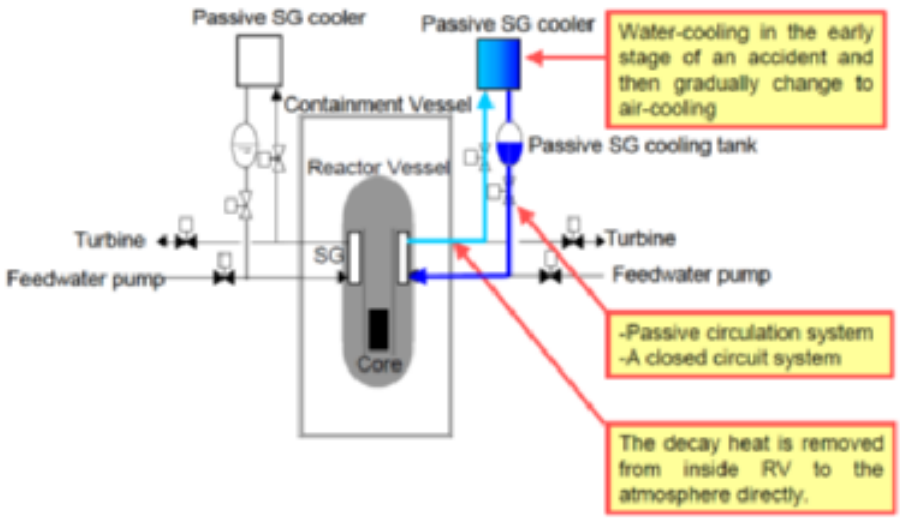


Figure 3: Stand-alone Direct Heat Removal System of IMR
(Reproduced courtesy of Mitsubishi Heavy Industries, Ltd.)

Containment System

A small containment vessel (CV) is made possible in the IMR by adopting an integrated primary system design and simplified systems. The IMR adopts reinforced concrete containment. The design pressure of the containment is planned to be 9 MPa to meet the safety requirement that water leakage from RPV will be terminated automatically. Since this CV is about one size larger than the RPV, it will be able to resist high pressure. The containment is part of the reactor containment facility. The reactor containment facility is part of the engineered safety systems, which include SDHS. The containment system is designed to suppress or prevent the possible dispersion of large quantities of radioactive materials, which would be released if extensive fuel failures were to occur in the reactor resulting from damage or failure of the reactor facilities such as the primary cooling system, main steam system, and feed water system.

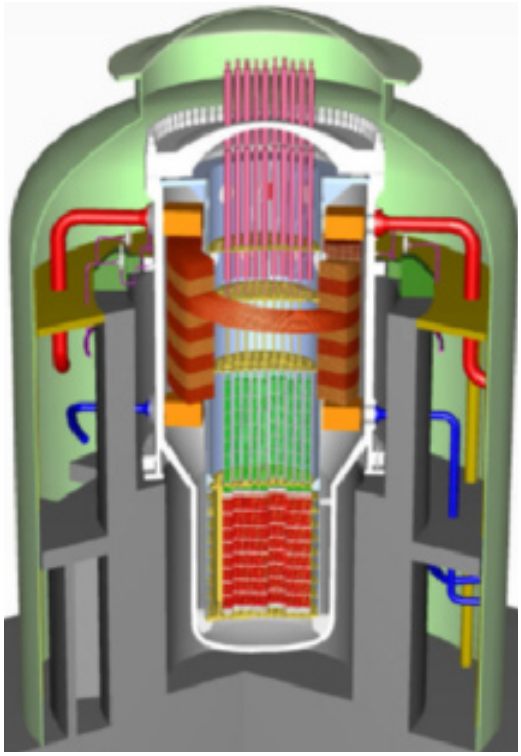


Figure 4: Compact primary containment vessel of IMR
(Reproduced courtesy of Mitsubishi Heavy Industries, Ltd.)

6. Plant safety and Operational Performances

The IMR reactor is designed to be operated automatically within the range of 20 to 100% of rated output power by the reactor control system. Even in the low output range below 20%, the control system can control the reactor automatically in the low power-operating mode. Usually, the primary system pressure and reactor power are controlled feedwater and control rods. When the load to the plant is changed, the feedwater flow rate and control rods are controlled simultaneously to steadily maintain the plant. The reactor control system is designed so that it can allow the following load demand change without causing a reactor trip:

- A +/- 10% step load demand change (within the range of 20 to 100%);
- A +/- 5% per min ramp load demand change (within the range of 20 to 100%);
- 100% load rejection.

7. Plant Arrangement

The concept of building layout and plant construction method is established, aiming at reducing the quantity of construction material, shortening the construction period and standardizing the plant design. The reduction in construction cost is established by applying steel plate reinforced concrete structure, simplifying the shape of building and the building structure.

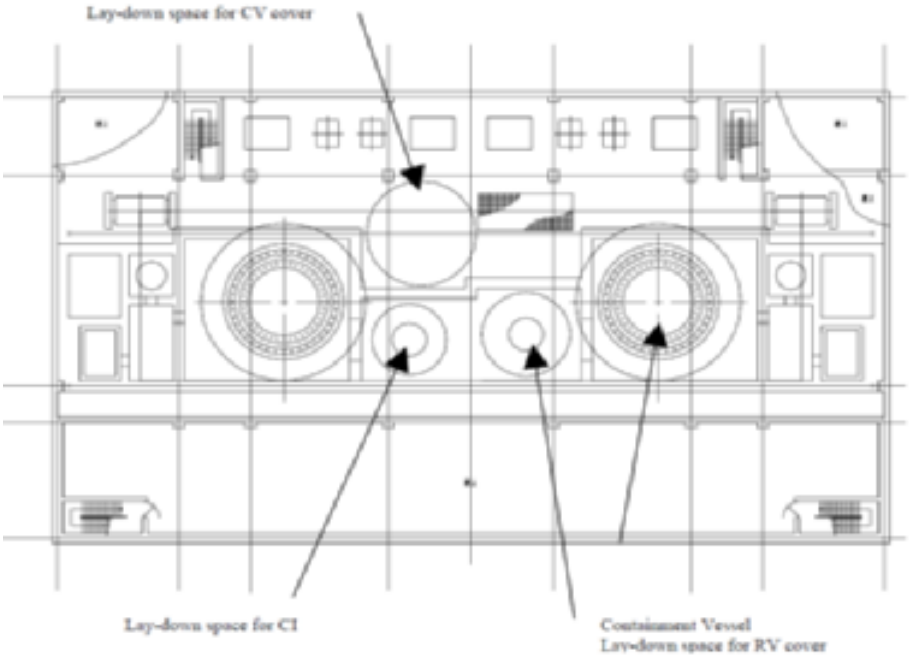


Figure 5: IMR Plant Layout Arrangement
(Reproduced courtesy of Mitsubishi Heavy Industries, Ltd.)

Reactor building
Ground level is assumed at above sea level, and flat land is assumed. The bedrock is assumed to be less than 40 meters below ground to enable the use of pile foundations. The reactor building is adopted in an isolated building and the turbine building is not adopted. The reactor building configuration is as follows:

- Integrated building consisting of two units;
- Exclusion of waste disposal facilities in another building;
- Adoption of steel structures in the building considering the module method.

Balance of plant:
The advanced BOP system allows the utilization of produced heat for non-electrical applications such as process heat, mining (oil sand extraction) and desalination.

Turbine Generator building
The turbine generator, turbine, condenser, moisture separator and reheater (MSR) and their auxiliary equipment are installed in the turbine building. The turbine generator is arranged with its axis in line with the reactor. Suitable spaces have been provided for inspection access, the transportation of tools for inspections and maintenance, and disassembly in such a way that the volume of the building is reduced.

8. Design and Licensing Status

At the moment, there is no domestic license or pre-license activities for SMR in general, since there is no SMR construction project in Japan. 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)

1. Introduction

The System-integrated Modular Advanced ReacTor (SMART) is an integral PWR with a rated electrical power of 100 MW(e) from 330 MW(th). SMART adopts advanced design features to enhance the safety, reliability and economics. The advanced design features and technologies implemented in the SMART were verified and validated during the standard design approval review. To enhance safety and reliability, the design configuration incorporates 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. The advanced design features of SMART include integrated steam generators (SG), in-vessel pressurizer and the horizontally mounted canned motor pumps. *Figure 1* shows SMART reactor system configuration.

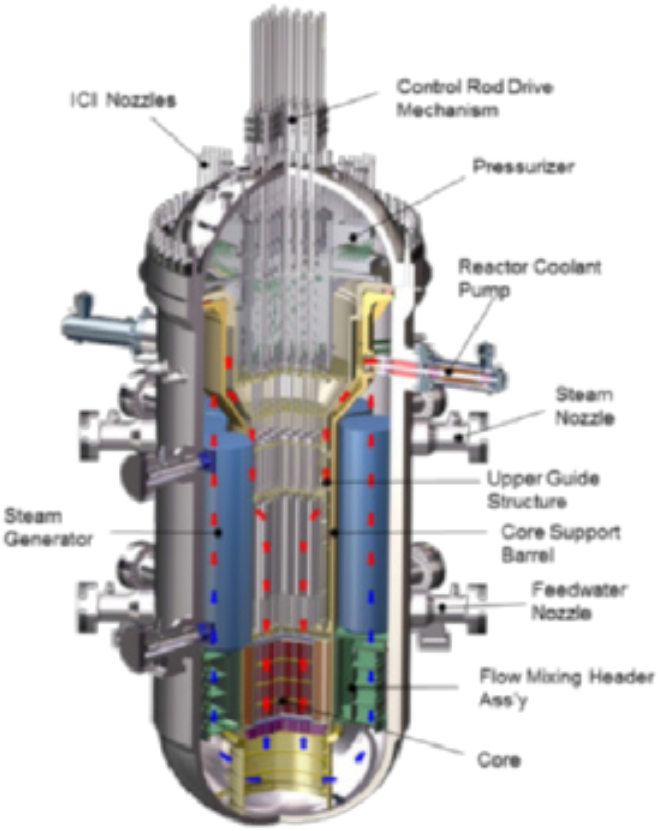


Figure 1: Reactor system configuration of SMART (Reproduced courtesy of KAERI)

2. Target Applications

SMART is a multi-purpose application reactor for electricity production, sea water desalination, district heating, process heat for industries and suitable for small or isolated grids. SMART has a unit output enough to meet the demands of electricity and fresh water for a city of 100000 populations. The SMART multi-purpose application configuration is shown in *Figure 2*.

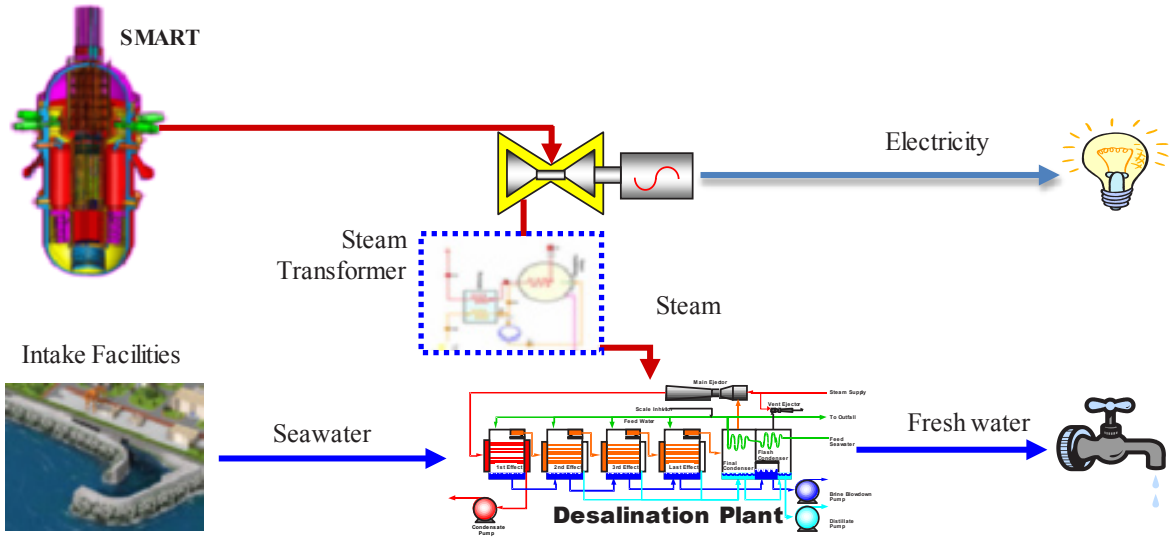


Figure 2: SMART multi-purpose application configuration (Reproduced courtesy of KAERI)

3. Development Milestones

March 1999	Conceptual design development
March 2002	Basic design development
June 2007	SMART-PPS (Pre-Project Service)
July 2012	Technology verification, Standard Design Approval (SDA)
March 2012	First step of Post-Fukushima corrections and commercialization
September 2015	Pre-project engineering agreement signed between Republic of Korea and Kingdom of Saudi Arabia for the deployment of SMART in the Gulf country

4. General Design Description

Design Philosophy

SMART design adopts an integrated primary system, modularization and advanced passive safety system to improve the safety, reliability as well as the economics. Safety performance of SMART is assured by adopting passive safety systems together with severe accident mitigation features. The passive safety features rely on gravity and natural circulation and require no active controls and operator intervention to cope with malfunctions and safety events. Improvement in the economics is achieved through a system simplification, in-factory fabrication, reduction of construction time and high plant availability.

Nuclear Steam Supply System

SMART has an integral reactor coolant system configuration that enables the elimination of large bore pipe connections resulting in the removal of the large break loss of coolant accident (LOCA) from the design bases events. The nuclear steam supply system (NSSS) consists of the reactor core, helically coiled steam generators and a pressurizer integrated in the reactor pressure vessel (RPV). The primary cooling system is based on forced circulation during normal operation. The system has natural circulation capability for emergency condition.

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 Availability Factor (%)	> 90
Design Life (years)	60
Plant Footprint (m^2)	90000
Coolant/moderator	Light water
Primary circulation	Forced circulation
Number of reactor coolant pump	4 (horizontally mounted on the reactor vessel)
System pressure (MPa)	15
Core inlet/exit temperatures (^oC)	296/323
Main Reactivity Control Mechanism	Control rod driving mechanisms and soluble boron
RPV Height (m)	18.5
RPV Diameter (m)	6.5
Module Weight (metric ton)	1070 (including coolant)
Configuration of Reactor Coolant System	Integrated
Power Conversion Process	Indirect Rankine Cycle
Fuel Type/Assembly Array	UO_2 pellet/17x17 square
Fuel Assembly Active Length (m)	2
Number of Fuel Assemblies	57
Fuel Enrichment (%)	< 5
Fuel Burnup (GWd/ton)	< 60
Fuel Cycle (months)	36
Cogeneration Capability	Yes
Approach to Engineered Safety Systems	Passive
Number of safety trains	4
Refuelling Outage (days)	45
Distinguishing features	Coupling with desalination and process heat application, integrated primary system, passive safety system.
Modules per Plant	1
Target Construction Duration (months)	36
Seismic design	> 0.18 g automatic shutdown
Predicted core damage frequency per reactor year	2E-7(internal events)
Design Status	Licensed/Certified (Standard Design Approval)

Reactor Core

The low power density design with a slightly enriched UO_2 fuelled core ensures a thermal margin of greater than 15%, which can accommodate any anticipated transient event. This feature ensures the core thermal reliability under normal and accident conditions. The fuel assembly (FA) is 2 m long and 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 two-batch refuelling scheme without reprocessing provides a cycle of 1000 effective full power days for 36 months of operation. This reload scheme minimizes complicated fuel shuffle schemes and enhances fuel utilization. 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.

Reactivity Control

Reactivity control during normal operation is achieved using control rods and soluble boron. Burnable poison rods are introduced for flat radial and axial power profile, which results in an increased thermal margin of the core. The number and concentration of the burnable absorber rods in each fuel type are selected so that reactivity of each assembly can be as flat as possible. SMART adopts typical magnetic-jack type control rod drive mechanism which has been widely used in the commercial nuclear power plants (NPPs). The small number of fuel assembly in the SMART assures a relatively high control rod worth.

Reactor Pressure Vessel and Internals

SMART integrated design means that the RPV contains all of the primary components. The RPV houses the core structures, eight (8) helical once-through steam generators (SGs), internal steam pressurizer and four (4) canned motor reactor coolant pumps.

Reactor Coolant System

SMART primary cooling system is based on forced circulation. The cooling of the reactor core is achieved by the four (4) canned reactor coolant pumps (RCPs) horizontally mounted at the upper shell of the RPV that circulate the coolant from the core to the SGs. Each RCP is an integral unit consisting of a canned asynchronous three phase motor and a mixed-flow, single-stage pump. The RCPs actively recirculate the coolant routing it into the core after having transferred heat to the secondary side of the SGs. The canned motor RCPs inherently eliminate potential small break LOCA associated with a pump seal failure which is one of the design basis events for conventional reactors. The reactor coolant forced by RCPs flows upward through the core and enters the shell side of the SG at the top of the SG. The secondary side feedwater enters the helically coiled tube side at the bottom of the SG and flows upward to remove the heat from the shell side eventually exiting the SG in a superheated steam condition. A large volume of the primary coolant provides large thermal inertia and a long response time, and thus enhances the resistance to system transients and accidents. These features ensure core thermal reliability under normal operation and any design basis events.

Steam Generator

The SMART has eight (8) modular type once-through SGs with helically coiled tubes to produce superheated steam under normal operating conditions. The SGs are located at the circumferential periphery between the core support barrel and RPV above the core to provide a driving force for natural circulation flow. The small inventory of the secondary side (tube side) water in each SG prohibits a return to power following a steam line break accident. In the case of accidents, the SG can be used as the heat exchanger for passive residual heat removal system (PRHRS), which permits an independent operation of the PRHRS regardless of the hydraulic condition of the primary system.

Pressurizer

The in-vessel pressurizer is designed to control the system pressure at nearly constant level for the overall plant operation. The large free volume formed in the upper part of the RPV is

used as the pressurizer region. The primary system pressure is maintained nearly constant due to the large pressurizer steam volume and the heater control. As the volume of the pressurizer is designed sufficiently large, condensing spray is not required for the load manoeuvring operation. The reactor over-pressure at the postulated design basis accidents related with a control failure can be reduced through the actuation of the pressurizer safety valve (PSV).

5. Safety Features

Safety systems of SMART are designed to function automatically on demand. These consist of a reactor shutdown system, a passive safety injection system (PSIS), PRHRS, a shutdown cooling system and a passive containment cooling system (PCCS). Additional safety systems include reactor overpressure protection system such as automatic depressurization system (ADS) and pressurizer safety valves, and severe accident mitigation system. As a result the reactor safety is enhanced substantially and the core damage frequency is also reduced.

Engineered Safety System Approach and Configuration

SMART has passive safety features to cope with abnormal operating occurrences and postulated design basis accidents, and severe accidents with core melts. The passive safety system is being developed to maintain a safe shutdown condition following design basis accidents such as LOCA and non-LOCA transient events without AC power or operator actions. The passive safety system consists mainly of PSIS, PRHRS, PCCS, and automatic depressurization system (ADS). 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. Engineered safety system of the SMART is shown in *Figure 3*.

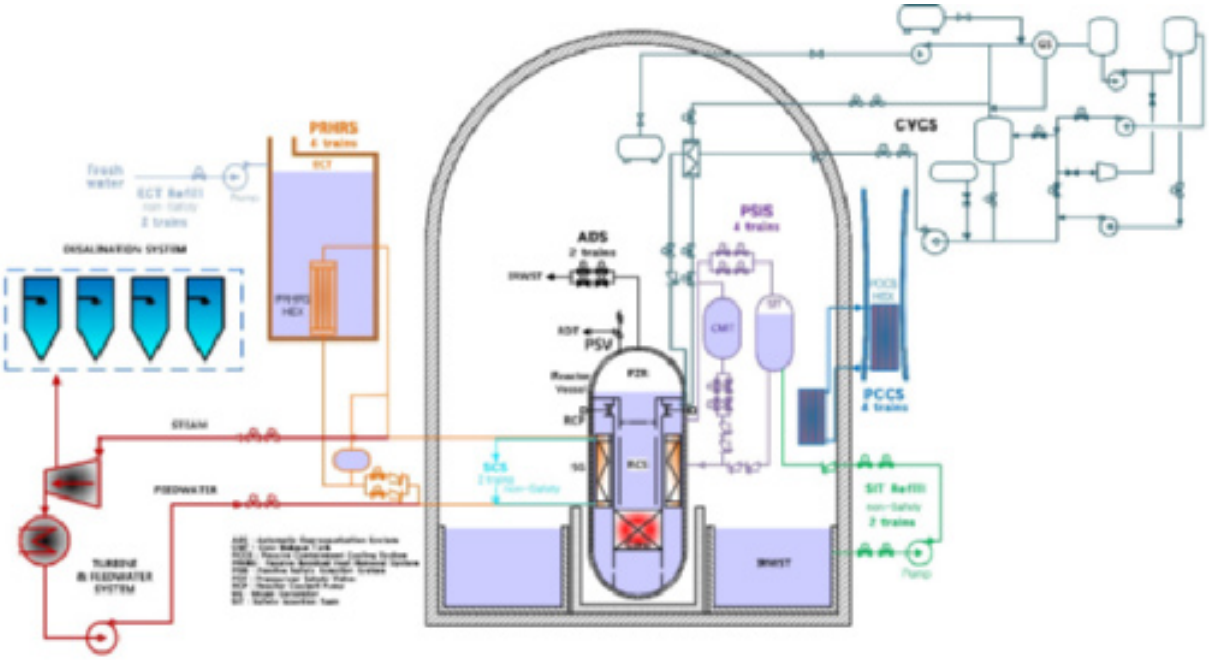


Figure 3: Engineered safety system of SMART (Reproduced courtesy of KAERI)

Decay Heat Removal System

After the reactor is shutdown, when the normal decay heat removal mechanism utilizing the secondary system is not operable by any reason, the PRHRS brings the RCS to the safe shutdown condition within 36 hours after accident initiation and maintains the safe shutdown condition for at least another 36 hours. Therefore the safety function operates for 72 hours

without any corrective action by operator for the postulated design basis accidents. The safety function of PRHRS is maintained continuously for a long term period when the emergency cooling tank (ECT) is replenished periodically. The PRHRS consists of four independent trains with a 33% capacity each, and each train is composed of an ECT, a heat exchanger and a makeup tank. In the design of PRHRS the possibility of loss of one train by a single failure is eliminated. Each train of PRHRS has a pair of check valves and isolation valves, both of which are installed on parallel lines. Therefore even if one valve fails, the whole train is still in operation. To remove the possibility of common mode failure, diversity of actuator is provided. Two kinds of different isolation valves are adopted, which are air-operated and electro-hydraulic valves. Therefore, single failure is no longer an issue in the design of PRHRS and three out-of-four trains are enough to remove the residual heat after an accident occurs.

Emergency Core Cooling System

Passive safety injection system (SIS) provides emergency core cooling following postulated design basis accidents. Emergency core cooling is performed using the four (4) core make-up tanks (CMTs) and four (4) safety injection tanks (SITs). Core cooling inventory is maintained through passive safety injection of CMTs and SITs with a 33% capacity each. The four (4) CMTs which are full of borated water provide makeup and borating functions to the RCS during early stage of SBLOCA or non-LOCA event. The top and bottom of CMT are connected to the RCS through the pressure balance line and the safety injection line, respectively. Each safety injection line is isolated by 2 x 2 parallel closed valves, which meets single failure criteria. Each pressure balance line is normally open to maintain pressure of the CMT at RCS condition. This arrangement enables the CMT to inject water to the RCS by gravity when the isolation valves are open. The isolation valves of the CMT injection line are actuated to open by low pressurizer pressure signal, high containment pressure signal, or PRHRS actuation signal. The four (4) SITs are filled with borated water and nitrogen gas at atmospheric pressure. These tanks provide makeup water to the RCS at low pressure conditions for about 72 hours in case of LOCA. The arrangement of the SIT is similar to that of CMT except that the isolation valves are installed at the pressure balancing line. The isolation valves of the SIT injection line are open when the RCS pressure reaches the set point pressure.

Containment System

The containment system is designed to contain the radioactive fission products within the containment building and to protect the environment against primary coolant leakage. The cylindrical primary containment system has a diameter of 44 m and a design pressure of 0.34 MPa. It protects the entire reactor systems from any external air-borne collisions. The containment system includes twelve (12) passive autocatalytic hydrogen recombiners and four (4) trains of passive containment cooling system (PCCS). The PCCS reduces containment pressure and temperature in case of main steam line break (MSLB) or LOCA, and removes fission products from the containment atmosphere following LOCA. The PCCS consists of four (4) independent trains with a 33% capacity each, and each train is composed of a heat exchanger, connecting pipe and valves. The PCCS dissipates the heat from the containment atmosphere to the environment through the heat exchanger by natural circulation.

6. Plant safety and Operational Performances

The load follow operation of SMART is simpler than that of large PWR because only the single bank movement and small insertion limit is required. SMART is suitable for load follow operation because of the small reactivity change for the power change due to the minimized coolant temperature change, relatively high lead bank worth due to a small number of fuel assemblies and the short effective core height leading to rapid damping of xenon oscillation. Daily load follow capability of SMART core performed shows that radial peaking factor, 3-dimensional peaking factor and the axial offset were satisfied within design limit. The load follow operation can be performed by the lead bank movement only within bank insertion limit without the change of soluble boron.

7. Instrumentation and Control Systems

High reliability and performance of I&C systems is achieved using advanced features such as digital signal processing, remote multiplexing, signal validation and fault diagnostics, and sensing signal sharing for protection & control system. Ex-core neutron flux monitoring system consists of safety and start-up channel detectors which are located within the RPV, and digital signal processing electronics. The signals from safety channels of the ex-core neutron flux monitoring system are shared to use for reactivity control system. In–core instrumentation system consists of 29 detector assemblies which are developed as mini type for SMART with four stacked rhodium self–powered detectors, one background detector and one set of K type thermocouple for each.

8. Plant Arrangement

Plant main building consists of the reactor containment building, the auxiliary building, the compound building (CPB), the emergency diesel generator building and the turbine-generator building (TGB). The CPB consists of an access control area, a radwaste treatment area, and a hot machine shop. For efficient radiation management, the plant main building is sub-divided into two zones; the duty zone and the clean zone.

Reactor Building

The reactor building and fuel storage area are equipped with a full monitoring system with closed circuit monitoring system (CCTV) to oversee and prevent unauthorized access to the fuel. Reactor building is a pre-stressed concrete shell structure composed of a right cylinder with a hemispherical dome, with steel plate lining to act as a leak tight membrane. Reactor building is erected on a common base-mat together with the auxiliary building in which the main control room and fuel storage area are located.

Main Control Room

The SMART compact control room is designed for one man operation under normal conditions of the plant. The main control room (MCR) is a key facility to cope with any emergency situations, so it is designed to ensure that plant personnel successfully perform the tasks according to the proper operating procedures. To achieve these goals, human factors engineering (HFE) process and principles are applied and verified using the full scope dynamic mock up. The four types of compact workstations such as safe shutdown monitoring and control workstation, which also contains a set of hardwired switches for manual actuation of ESF equipment and reactor trip, main monitoring and control workstation, auxiliary monitoring and control workstation, minimum inventory control workstation and a large display panel are installed in the MCR.

Figure 4: SMART general plant arrangement with a single and a twin unit (Reproduced courtesy of KAERI)

Balance of Plant:

Turbine Generator Building

The reference concept of the turbine plant has been developed including a coupling system for seawater desalination. The secondary system receives superheated steam from the NSSS. It uses most of the steam for electricity generation and preheaters, and the remainder for non-electric applications. Sea water desalination system may be used in conjunction with the secondary system. The steam transformer produces the motive steam using steam extracted from a turbine and supplies it to the desalination plant.

Electric Power Systems

These systems include the main generator, main transformer, unit auxiliary transformers, stand-by auxiliary transformers, diesel generators, and batteries. The electrical systems, including the class 1E and non-class 1E, are based on a two train approach. The main generator is connected to the grid via the main transformer with auxiliary transformers connected between them. The unit auxiliary and/or the stand-by transformers supply electrical power for start-up, normal operation and shutdown.

9. Design and Licensing status

Korea Atomic Energy Research Institute (KAERI) received the standard design approval from the country's nuclear safety commission in July 2012. In September 2015, a pre-project engineering agreement was signed between the Republic of Korea and the Kingdom of Saudi Arabia for deployment of SMART. A construction period of less than three (3) years from first concrete to fuel load is predicted. A safety enhancement program to adopt passive safety system in SMART began in March 2012, and the testing and verification of the PRHRS and PSIS were completed in the end of 2015.

10. Plant Economics

The target plant construction cost is $10000/kWe and an operating & maintenance cost of 4.5¢/kWh. For a twin unit of SMART, the total cost is expected to be 20% lesser.

UNITHERM (NIKIET, Russian Federation)

1. Introduction

The UNITHERM is a small transportable nuclear power plant (NPP) with a capacity of 30 MW(th) and a rated electrical output of 6.6 MW. UNITHERM is developed based upon NIKIET's experience in designing marine nuclear installations. The UNITHERM reactor is intended for electricity supply to urban areas and industrial enterprises in remote regions. UNITHERM adopts a natural circulated primary cooling system and is intended for minimal operational staffing with an option for unattended operation and a centralized regional support facilities monitoring. The UNITHERM design adopts proven technology and operational experience of the WWER type reactors. The design aims for fabrication, assembly and commissioning of the NPP modules to be carried out at factory. The UNITHERM reactor is designed to operate for 20-25 years without refuelling. The UNITHERM reactor design allows for its use both as a land-based and barge mounted NPP. NPP with UNITHERM may consist of a number of units depending on the purpose and demand of costumers need.

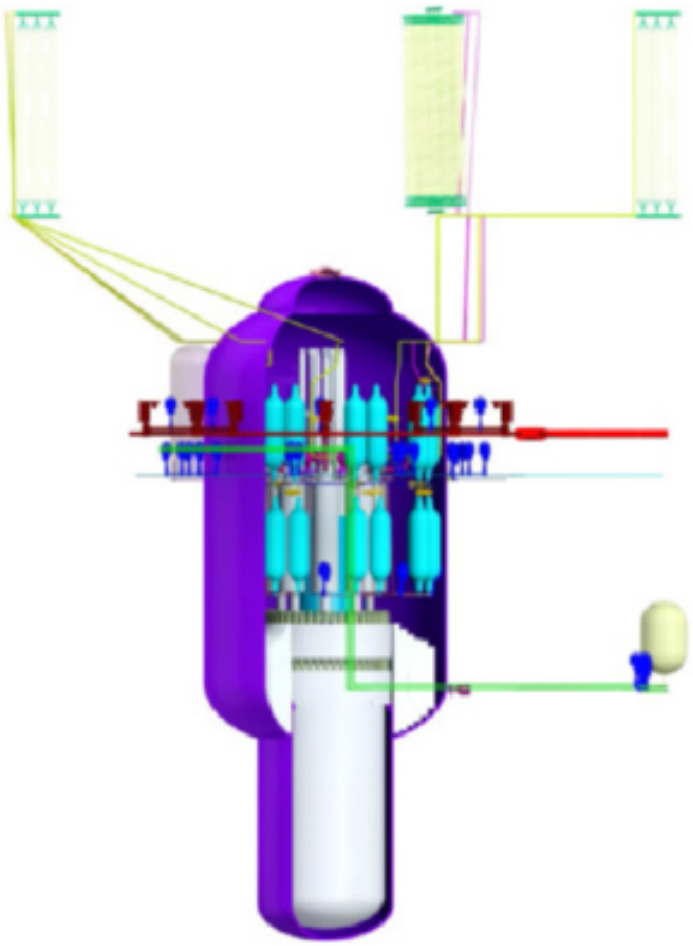


Figure 1: Reactor System Configuration of UNITHERM
(Reproduced courtesy of NIKIET)

2. Target Application

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, the 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 design.

3. Development Milestones

1994	The NPP design on the basis of the UNITHERM concept has become the laureate of the competition on SMR designs established by the Russian Nuclear Society
2012	Technical proposal on the UNITHERM reactor facility (WDR stage)
2015	Technical proposal for a SMR plant based on the UNITHERM reactor

4. General Design Description

The UNITHERM plant is designed to operate without presence of operating personnel. The design aims to provide the reactor maintenance in routine and urgent cases from regional centres common to several plants of this kind. The design adopts inherent safety features which include: large coolant inventory, negative reactivity coefficients and a high flow rate of coolant in natural circulation. One of the main features of the UNITHERM is the suitability for shop fabrication and testing.

Design Philosophy

NPPs with the UNITHERM reactor are designed for siting in remote regions with less developed infrastructure and where qualified staff for plant operation may not be available. The reactor core life is expected to be equal to the plant lifetime with an estimated time of 20-25 years. The refuelling of the core will not be required during the plant service life.

Nuclear Steam Supply System

Primary circuit system is intended for heat removal from the reactor core and heat transfer to the intermediate circuit fluid inside the intermediate heat exchanger. The system consists of a main circulation train and a pressurizing system. The natural circulation of primary coolant takes place in the primary circuit. The intermediate circuit system is intended for heat transfer from the intermediate circuit coolant to the secondary coolant (consumer's circuit) inside a steam generator (SG). This system provides an additional localizing safety barrier to protect the heat consumers against the ionizing radiation from radionuclides generated by primary coolant activation, from structural material corrosion products dissolved in the primary coolant as well as fission products entering the primary circuit in case of fuel cladding failure. Primary coolant circulates by means of natural convection. The secondary system (consumer's circuit) is intended to generate a superheated steam from the feedwater (supplied by NPP turbogenerator pumps) by means of heat transfer from the intermediate circuit coolant inside the steam generator. Secondary coolant circulates by means of natural convection.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	NIKIET, Research and Development Institute of Power
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
Core inlet/exit temperatures (^oC)	249 / 330
Main reactivity control mechanism	Soluble boron and control rod insertion
RPV height (m)	9.8
RPV diameter (m)	2.9
RPV or module weight (metric ton)	180 (integral reactor module)
Configuration of reactor coolant system	Integral type
Power conversion process	Direct Rankine Cycle
Fuel type/assembly array	UO_2 particles in a metallic (silumin or zirconium) matrix , metal-ceramic/ 54-55
Fuel assembly active length (m)	1.1
Number of fuel assemblies	265
Fuel enrichment (%)	19.75
Fuel burnup (GWd/ton)	1.15
Fuel cycle (months)	200
Cogeneration capability	Yes (heat mode power: 2.5 MW(e), 20 MW(th) heat)
Approach to engineered safety systems	Hybrid (Passive + Active) system
Number of safety trains	2
Refuelling outage (days)	45-50
Distinguishing features	Permanently operational autonomous passive system for abstract of heat from reactor; guard vessel; iron-water biological protection of reactor; passive systems for abstract of heat from the guard vessel and biological shielding tanks
Modules per plant	As per customer requirement
Target construction duration (months)	84
Seismic design	VIII-IX-MSK 64
Core damage frequency (per reactor year)	1E–6
Design Status	Conceptual design

Reactor Core

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 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. A specific feature of the UNITHERM fuel cycle is the long and uninterrupted irradiation of fuel inside the reactor core throughout the reactor lifetime, without 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.

Reactivity Control

The control element drive mechanisms (CEDMs) are designed to provide secure insertion of rods in the core by gravity for reactivity control. Locking devices are installed in the CEDM to avoid unauthorized withdrawal of control rods. Burnable absorbers are used to compensate the decrease of reactivity due to fuel burn-up, temperature effect and by motion of the reactivity control rods during periodic maintenance.

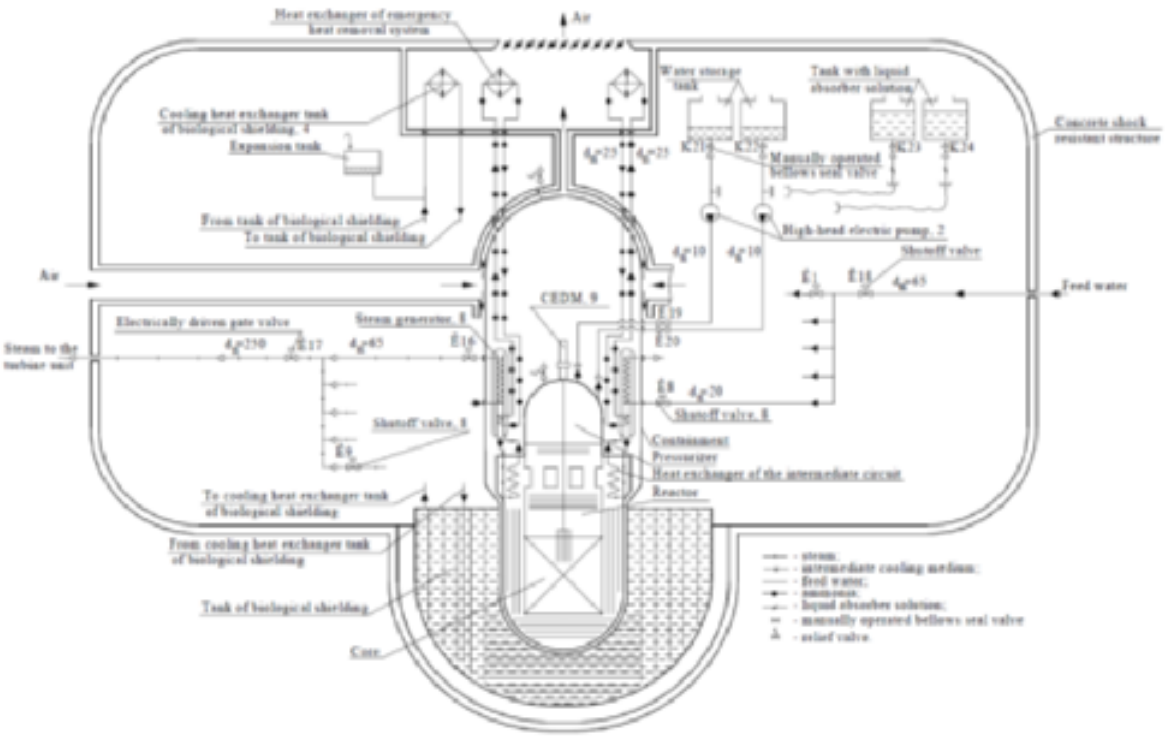


Figure 2: UNITHERM schematic diagram
(Reproduced courtesy of NIKIET)

Reactor Pressure Vessel and Internals

UNITHERM is an integral type reactor with nuclear steam supply system (NSSS) equipment installed inside the reactor pressure vessel (RPV).

Reactor Coolant System
The UNITHERM primary cooling mechanism under normal operating condition and shutdown condition is by natural circulation of coolant. The heat energy released from the reactor core is transferred to the intermediate circuit coolant, which moves upward to flow outside the tubes of the helically coiled once-through steam generator (SG).

Steam Generator
The reactor employs a helically coiled once-through SG. Heat transfer from the reactor core to the intermediate circuit coolant occurs in the built-in once-through intermediate heat exchanger and heat transfer from the intermediate coolant to the consumer's circuit coolant inside the SG. Both heat exchangers are made from titanium alloy. The intermediate heat exchanger has a structure of coil bundle consisting of 80 separate subsections that are united in 8 independent sections. Their supply and discharge tubes are connected to 8 pressure vessel steam generating modules installed on the reactor cover.

Pressurizer
Pressurizer of UNITHERM is a built-in structure of the upper plenum of the RPV.

5. Safety Features

The UNITHERM safety philosophy is to assure that the radiation impact on personnel, population and the environment under normal and design basis accidents is well below the limits prescribed by the current regulations. The UNITHERM design makes use of passive systems and devices based on natural processes without external energy supply. The design inherently eliminates potential hazardous activities related to the core refuelling, as the reactor core refuelling will not be required in the plant service life. This further simplifies the operating technologies and enhances the proliferation resistance.

Engineered Safety System Approach and Configuration
The UNITHERM safety systems are based upon redundancy, diversity and the maximum use of the fail-safe systems. The UNITHERM employs passive safety systems and devices which do not require actuation (such as containment, independent heat removal system, etc.) or can be passively actuated (such as primary circuit systems and containment depressurization system). The reliability and safety of the UNITHERM reactor is significantly improved due to the elimination of the shut-off and isolation valves from the reactor pipelines, except for the user circuit, i.e., all systems are in continuous operation. The component cooling circuit is passively operated and continuous removal of heat from the reactor components enclosed in the containment is achieved efficiently. The structures of the UNITHERM NPP are designed to protect the reactor from extreme external events such as hurricanes, tsunami, aircraft impacts, etc. The reactor can be automatically shut down and brought to a safe state without exceeding the design limit. The UNITHERM also incorporates several design features and measures for protection from human errors and mitigation of the consequences of human errors or acts of malevolent.

Emergency core cooling system
An independent passive heat removal system is adopted which acts as a cooldown system in emergency shutdown of the reactor. During a postulated loss of coolant accident (LOCA) scenario, some primary coolant and steam-gas mixture from the pressurizer are discharged to the containment. The emergency core protection system is activated in response to the signals from pressure transducers. Coolant leakage continues until the pressure values in the reactor and containment are equalized. The remaining coolant inventory in the reactor is sufficient to maintain circulation in the primary coolant circuit. The reactor is passively cooled via the

intermediate circuit and the independent heat removal circuit, whereas the containment heat is removed by the component cooling system. Additionally, an active user circuit with feedwater supplied to the SG and steam-water mixture maybe utilized to increase the cooling rate. The iron-water biological shield acts as a system of bubble tanks for cooling water storage. The shielding removes heat from the RPV, preventing a core melt in a postulated beyond design basis accident with reactor core voiding.

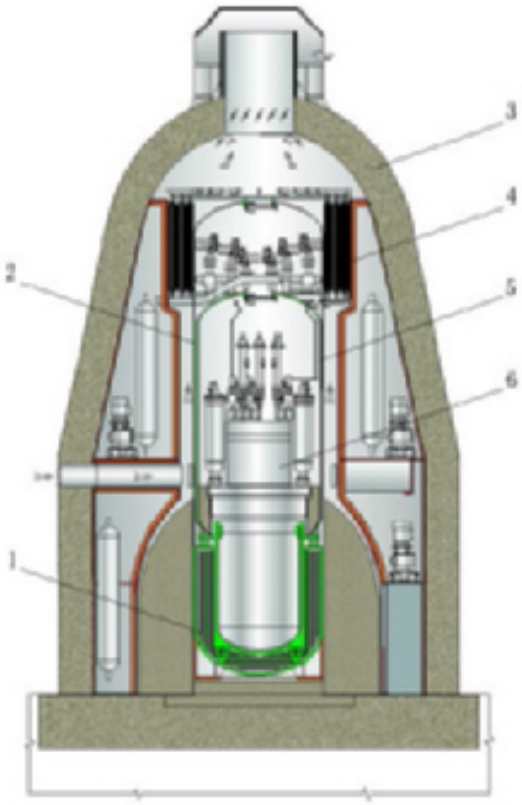


Figure 3: (1) iron-water shielding tank; (2) containment; (3) shock-proof casing; (4) cooldown system heat exchanger; (5) safeguard vessel; and (6) the reactor (Reproduced courtesy of NIKIET)

Containment system

The integral reactor for land-based deployment 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 containment system is capable of maintaining the primary coolant circulation as well as provides reactor cooldown and retention of radioactive products under the loss of primary circuit leak tightness. Passive safety systems for the removal of heat from the containment and biological shielding tanks are employed.

6. Plant Arrangement

Turbine Generator building

Turbine-generator assembly for UNITHERM NPP depends on the plant capacity and operation mode requested by its users. The turbine operates using dry saturated steam in the mode of steam outlet backpressure. With consideration of the continuous transfer of 5% heat to the independent heat removal system, the total efficiency in this case is expected to be ~74%. High efficiency is achieved from the utilization of low-parameter heat at the turbine

exhaust. An electric generator with an output of 6.6 MW(e) in combination with a single-phase intermediate circuit allows to obtain a superheated steam temperature of 285°C under 1.35 MPa.

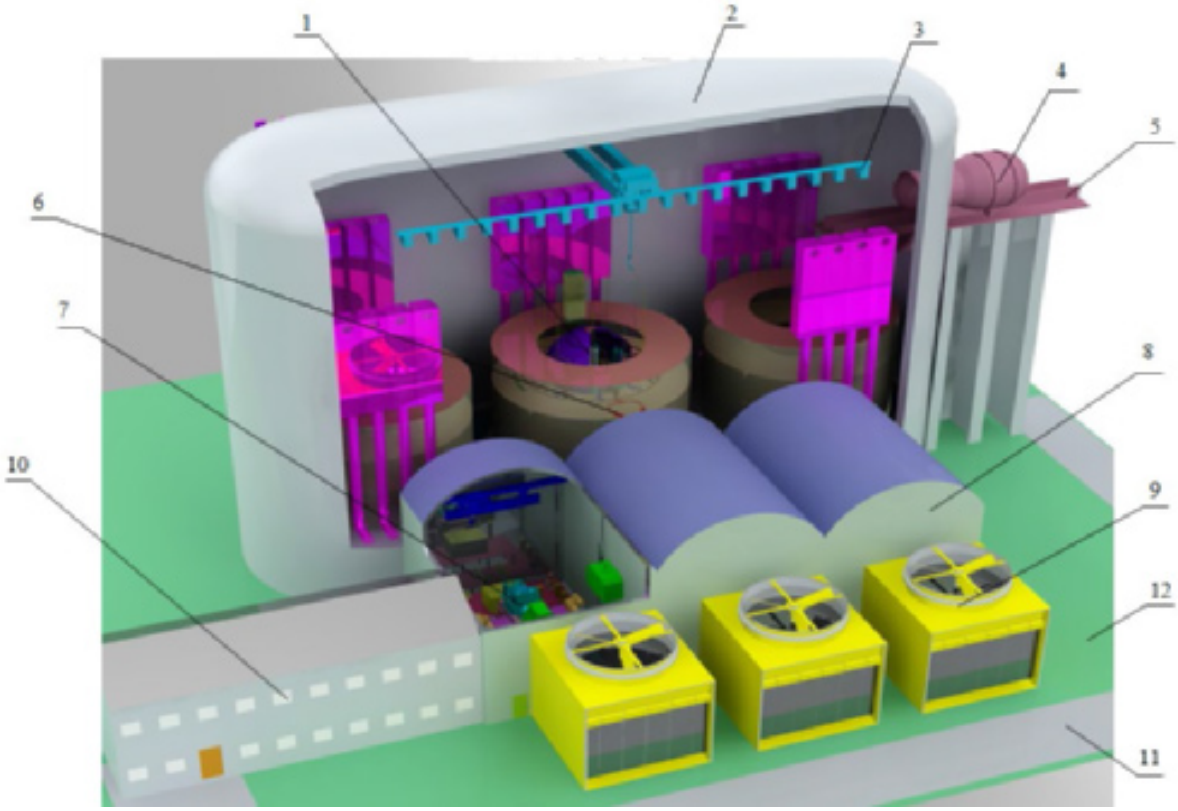


Figure 4: The UNITHERM plant arrangement (Reproduced courtesy of NIKIET)
1 – reactor plant; 2 – reactor building; 3 – bridge crane; 4 – transfer lock; 5 – overpass with the crane; 6 – pipelines and components inside the reactor building; 7 – turbine generator; 8 – turbine building; 9 – air cooler; 10 – administrative and amenity building; 11 – roads; 12 – site area

7. Design and Licensing Status

Based on the experience of NIKIET and other Russian institutions and enterprises in the development of marine nuclear installations, the UNITHERM NPP may require no major research and technology development activities for its deployment. Once an agreement with the user is reached and the technical assignment approved, it is estimated that 5 years will be required to finalize design development, licensing, construction and commissioning of the UNITHERM NPP, provided there are no financial or organizational constraints. The detailed design stage would include qualification of the core, heat exchangers, CEDMs and other components.

8. Plant Economics

The serial production of plant equipment fully under shop conditions, including equipment assembly and testing is expected to facilitate the low capital costs for plant construction

KARAT-45 (NIKIET, Russian Federation)

1. Introduction

KARAT-45 is a small boiling water reactor (BWR) with a rated power of 45 MW(e) designed by NIKIET as an independent cogeneration plant for producing electric power, steam and hot water. It is developed as the base facility for the economic and social development of the Arctic region and remote extreme northern areas of Russian Federation.

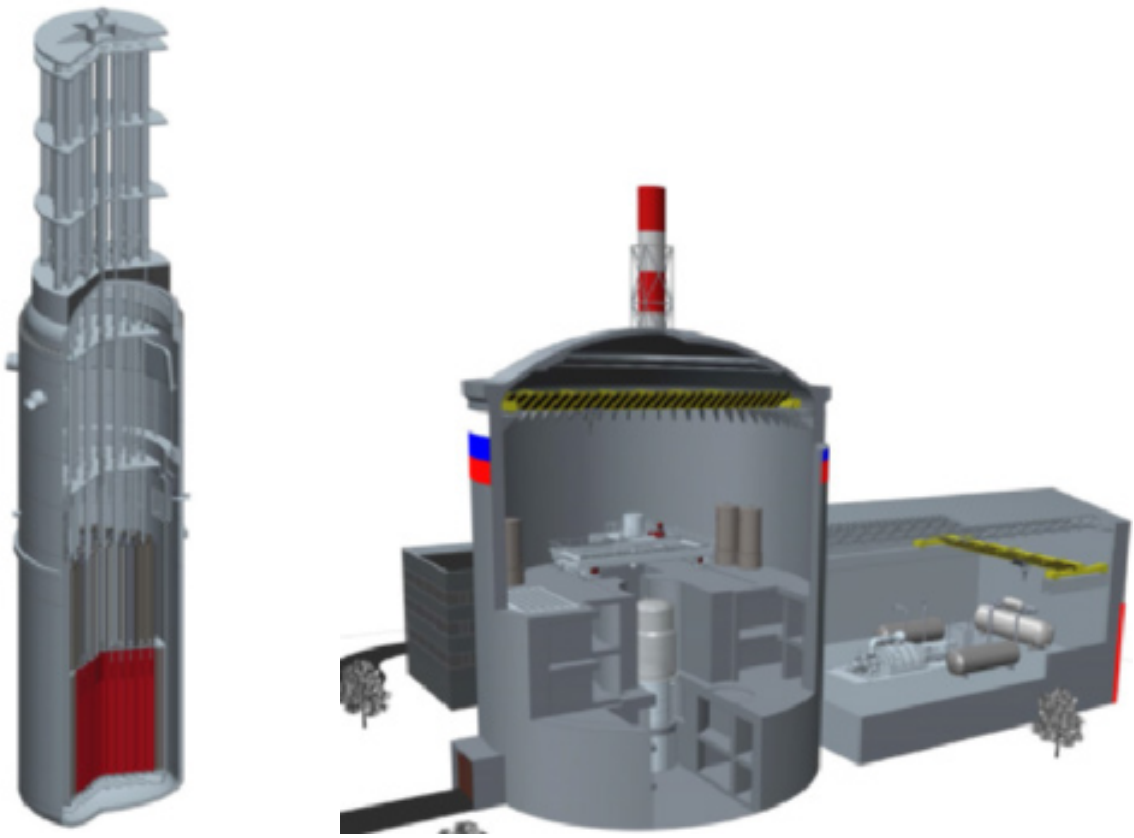


Figure 1: Reactor System Configuration of KARAT-45
(Reproduced courtesy of NIKIET)

2. Target Application

KARAT-45 is a small boiling water reactor (BWR) with a rated power of 45 MW(e) designed as an independent cogeneration plant for supply of electric power, steam heat and hot water, as shown in *Figure 2*. KARAT-45 power unit has a high load follow capability to cope with daily power variation from 20% to 100% of nominal capacity.

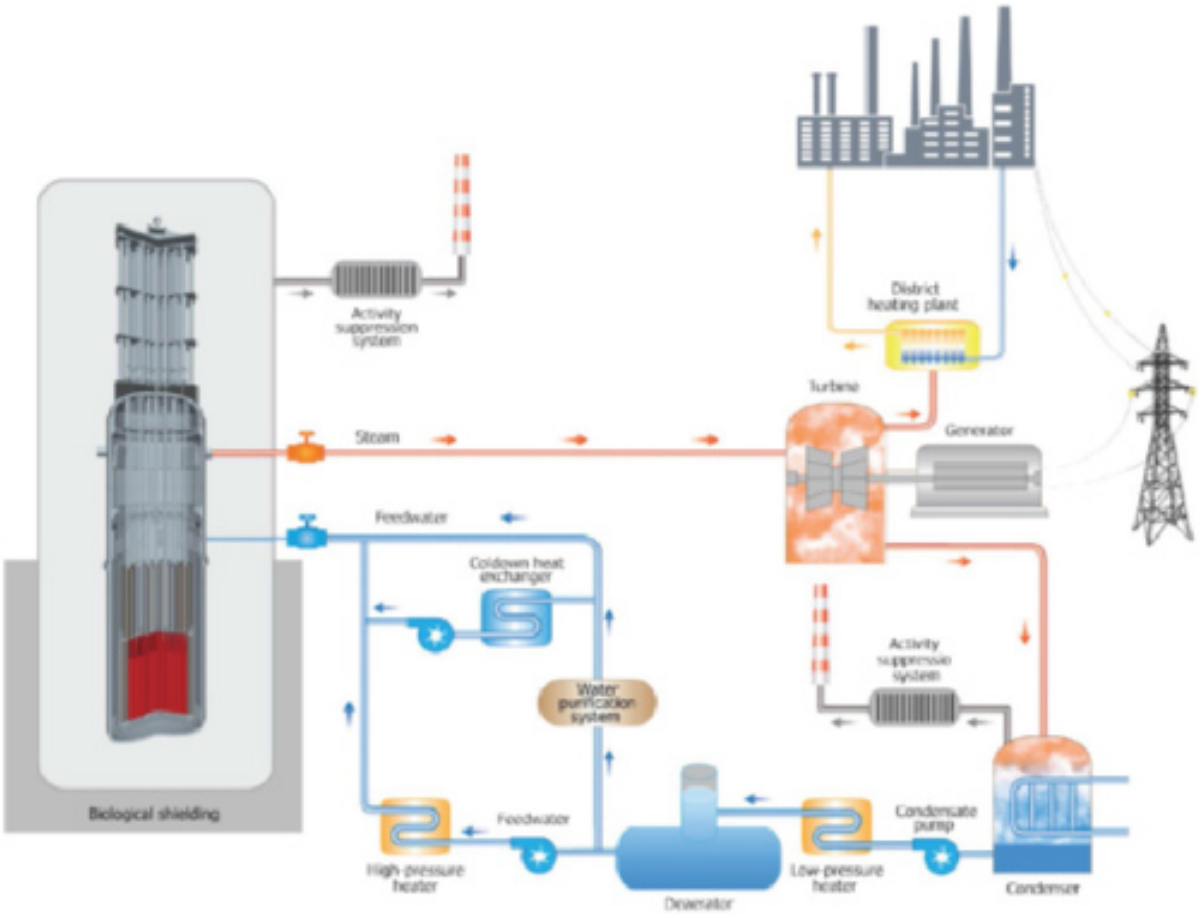


Figure 2: Application Scheme of KARAT-45
(Reproduced courtesy of NIKIET)

3. Development Milestones

2015	Development of a Detailed Project Report
2016	Project definition, technical requirements, R&D program
2017 – 2019	Basic design, R&D, development of PSAR, expert review and licensing
2020 – 2022	Detailed design, fabrication of equipment, construction
2023	Operation license, first criticality, first start, commissioning

4. General Design Description

Design Philosophy

The BWR technology was selected as a basis for the design and technology development of KARAT-45 due to the following rationales: BWR employs single circuit removal of heat so capital cost for construction can be minimized; lower system pressure poses fewer challenges to the reactor vessel; BWR has inherent self-protection and self-control properties due to negative void and temperature reactivity coefficients. KARAT-45 complies with Russian regulatory requirements and IAEA guidelines. The primary cooling mechanism for the reactor core is natural circulation for all operating modes. The reactor vessel because of its small size will be shop-fabricated in modular fashion to make it transportable. The reactor is designed for a long service life.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	NIKIET
Country of origin	Russian Federation
Reactor type	Boiling Water Reactor
Electrical capacity (MW(e))	45 – 50
Thermal capacity (MW(th))	180
Heat supply, base mode (Gcal/h)	45 – 50
Expected capacity factor (%)	92
Continuous operation period (hr)	19200
Design life (years)	80
Plant footprint (m^2)	9000
Coolant/moderator	Light water
Primary circulation	Natural circulation
System pressure(MPa)	7.0
Core inlet/exit temperatures (^oC)	180 / 286
Core flow rate (ton/hr)	2600
Steam generating capacity (ton/hr)	320
Steam temperature (^oC)	286
Pressure (MPa abs)	7.0
Main reactivity control mechanism	Control Rod Drive Mechanism (CRDM)
RPV height (m)	11.15
RPV diameter (m)	3.10
Module weight (metric ton)	250
Configuration of reactor coolant system	BWR
Power conversion process	Direct Rankine cycle
Fuel type/assembly array	UO_2 pellet/hexagonal
Fuel assembly active length (m)	1.98
Number of fuel assemblies	109
Fuel enrichment (%)	4.5
Fuel burnup (GWd/ton)	45.9
Fuel cycle (years)	7
Cogeneration capability	Yes
Approach to engineered safety systems	Passive
Number of safety trains	2
Refuelling outage (days)	30 (once in three years)
Distinguishing features	Designed for extreme Arctic and Northern area conditions
Modules per plant	1-4
Target construction duration (months)	48
Seismic design (g)	3g
Core damage frequency (per reactor-year)	< 1E-6
Design Status	Conceptual design

Nuclear Steam Supply System

The reactor employs a single-circuit heat removal system. The reactor steam removal system is designed to transport the steam generated in the reactor to the turbine. The system removes heat from the reactor during the reactor start-up, power operation and shutdown, as well as in some operational events when this system and the feed water supply system are serviceable. In normal conditions of the reactor power operation, saturated steam is fed from the reactor to the turbine via two steam lines. Isolation valves of the primary circuit leak proof enclosure formed by the primary containment are installed immediately in front of and behind the containment penetration. The isolation valves are opened during normal operation. KARAT-45 reactor's design is based on gravity-type steam separation without centrifugal axial separators, which adds to the reactor safety and makes it different from some other similar type reactors of bigger scale.

Reactor Core

The reactor core as shown in *Figure 3* is located in the lower part of the reactor and consists of 109 fuel assemblies (FA). The core has five complete FA rows and an incomplete sixth row. There are six steel reflector blocks at the core periphery for the vessel protection against radiation. The FAs are installed inside the support grid locations. In the upper part, the FAs are arranged in a hexagonal lattice with a pitch of 185 mm. Control orifices are installed at the core inlet for the coolant flow hydraulic profiling.

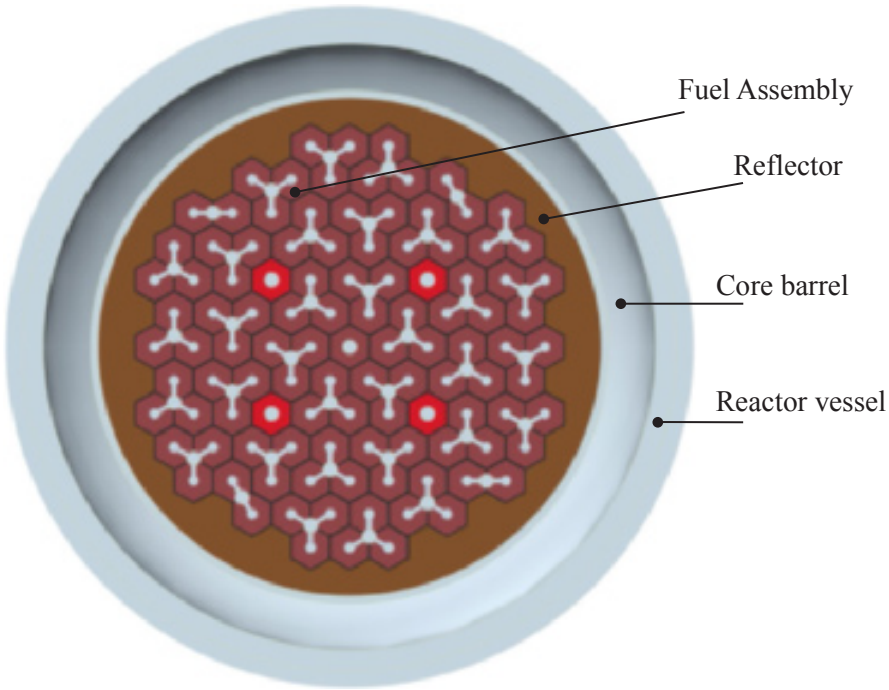


Figure 3: KARAT-45 reactor core (Reproduced courtesy of NIKIET)

Reactivity Control

The core includes 109 Control and Protection System (CPS) rods fitted in the FA guide channels. Functionally, the rods are divided into 4 emergency protection rods and 105 control rods. A rod is a shroud less structure consisting of eight cylindrically shaped absorber elements. The absorber material is boron carbide (B_4C). The control rods are grouped into clusters to reduce the number of actuators. One actuator is used to move three, two or one rod.

Reactor Pressure Vessel and Internals
The reactor pressure vessel (RPV) accommodates the reactor core and the reactor internals, including two feedwater supply headers, two emergency heat exchangers and a block of louver-type separators. The reactor vessel has an elliptical bottom, cylindrical shells, and two nozzle and main connector flange shells welded one to the other. The RPV outer diameter is 3100 mm, wall thickness – 100 mm, and height - 11150 mm. The functions of the reactor internals are:

- positioning and retention of the reactor core;
- arrangement of coolant circulation;
- steam separation;
- heat transfer from the core;
- vessel protection against neutron irradiation.

Reactor Coolant System
The reactor has a single-circuit heat removal system. The coolant circulation is natural. The coolant (light water) flows through the reactor core upwardly while being heated and boiled. The steam generated after drying is fed to the turbine before being discharged into the condenser downstream of the turbine. Some part of the steam could be extracted from the turbine for heating the plant's in-house water and for heat supply. After leaving the condenser, the water is pumped through heaters and enters the deaerator. Using feedwater pumps, the deaerated condensate is fed into the reactor through feedwater supply headers. Inside the reactor, the feedwater is mixed with leftover water from steam separation and fed to the core inlet.

Steam Generator
Steam is generated directly inside the reactor vessel and, having been dried in the steam space, is fed into the turbine with a humidity of not more than 0.1 wt.% using gravity separation and louver-type separators built in the reactor. Downstream of the turbine, the steam is dumped into the condenser.

Pressurizer
There is no pressurizer. Pressurization is achieved through negative feedbacks on temperature, power and void reactivity effects.

5. Safety Features

The safety concept of the KARAT-45 reactor is based on inherent self-protection features, the defence-in-depth approach and a system of barriers to the release of radioactive materials into the environment. The concept is aimed at preventing accidents and mitigating their consequences, should these occur. To achieve this, normal operation systems and safety systems are required to perform three major functions:

- reactivity control;
- core cooling;
- confinement of radioactive materials in the required limits.

Engineered Safety System Approach and Configuration
One of the major principles of safety systems design is the requirement that they should operate at any design-basis initiating event and during failure of any active or passive component with mechanical parts independently on the initiating event (single failure

principle). The safety system design also meets the requirement for the systems to perform its functions automatically and reliably with the smallest possible number of active elements involved and using the passive protection principle.

Decay heat removal system
The decay heat removal system is designed to remove heat from the reactor core during unexpected operational occurrences and events caused by a loss of heat removal due to the feedwater supply and steam discharge systems failure. The system ensures the nuclear fuel cooling function. The system is based on a passive principle of action with heat removed from the reactor through natural circulation.

Boron solution injection system
The boron solution injection system is designed to maintain the reactor subcritical while the power system and independent power supply are disconnected, including accidents involving coolant leakage. The criterion for the performance by the system of its function is that the reactor is maintained subcritical even in the event of CPS actuators failure and the reactor is cooled by water in pre-emergencies involving coolant leakage.

Emergency core cooling system
The emergency core cooling system is designed to supply the in-vessel natural circulation circuit with water during accidents with loss of the primary circuit integrity. The system uses passive principle of action to organize the coolant movement. The emergency mitigation of the primary coolant loss is ensured passively by draining water from the emergency cooldown tanks into the reactor due to the gravitation because of the difference in the tank and reactor elevations.

Containment system
KARAT-45 reactor is located inside a reinforced concrete containment with a stainless steel lining. The containment serves to localize accidents and is designed to withstand a pressure of up to 3 MPa. It forms an additional barrier to the leakage of radioactive materials into the environment while limiting, by its volume, the coolant loss during a reactor vessel break. There are isolation gate valves installed on pipelines at the containment outlet.

6. Plant safety and Operational Performances

The major objective of the safety assurance arrangements is to limit the KARAT-45 radiation impacts on the personnel, local population and the environment during normal operation, anticipated operational events and accidents. KARAT-45 reactor features the following inherent self-protection properties:

- negative temperature and void reactivity coefficients;
- passive cooling of the reactor core based on natural coolant circulation both during normal operation and anticipated operational violations;
- sufficient amount of water in the emergency cooldown tank (ECT) for long-term decay heat removal;
- moderate thermal power density of fuel elements and reliable removal of residual core heat by merely filling the core with coolant;
- a substantial amount of coolant above the reactor core to ensure the reliable fuel cooling in majority of possible emergencies;
- self-limitation of the in-vessel pressure variation rate due to the damping properties of the steam blanket.

7. Instrumentation and Control systems

In-core monitoring system is designed for monitoring of thermal-hydraulic and neutron properties of the reactor core and in-core coolant natural circulation flow which are measured directly or indirectly in different operating modes of the reactor. The following parameters are expected to be monitored in the KARAT-45 reactor:

- continuous monitoring of neutron parameters defining the reactor period, neutron power and the control rod position;
- continuous monitoring of thermal parameters defining the reactor's thermal power, the reactor water level, and temperature of water at FA inlet, in steam space and at the reactor vessel surface;
- periodic water chemistry control;
- periodic inspection of the thermal reliability of the reactor core operation. In-core power density field monitoring.

8. Plant Arrangement

The building layout plan for the land-based power unit of KARAT-45 is designed in such a way that the reactor system, including its servicing systems, spent fuel pool, and auxiliaries are located in double protective air-crash resistance buildings. The overall weight and size parameters of the reactor unit due to its modular nature and transportability allow the delivery of unit assembled at factory directly to the construction site by railway or other means of transportation.

Reactor building

The reactor unit building performs the function of a primary containment. The reactor unit accommodates KARAT-45 reactor, as well as systems responsible for emergency heat removal from the reactor, emergency reactor shutdown and removal of radiolysis products from beneath the reactor head. Besides, the reactor building includes an irradiated FA storage facility and its cooling system, as well as the reactor facility's moving and handling equipment.

Control building

The control building houses the main control room and the emergency control room, electric switchboard room, personnel rooms, communications center and other administrative rooms.

Balance of the plant:

Turbine Generator building

The turbine block accommodates the turbine generator, steam condensate line components and equipment, and a bridge crane for moving operations. The dimensions of the turbine block are 35x22 m, and its height is 28 m.

Electric power systems

The normal power supply system is designed to supply electric power to all plant consumers during normal operation and anticipated operational snags, including accidents, as well as to deliver the electricity generated by the turbine plant to offsite and in-house consumers.

9. Design and Licensing Status

KARAT-45 design was developed in conformity with Russian laws, norms and rules for NPPs and safety principles developed by the world community and IAEA recommendations.

KARAT-100 (NIKIET, Russian Federation)

1. Introduction

KARAT-100 is an integral type multi-purpose boiling water reactor (BWR) with a power output of 360 MW(th) and a rated electrical output of 100 MW(e). The design adopts engineering approaches proven at prototype and testing facilities. The reactor is designed for the production of electrical power, heat for district heating and hot water in cogeneration mode. The design adopts natural circulation for its primary cooling system core heat removal in all operational modes. The design configuration incorporates passive safety systems to enhance the safety and reliability.

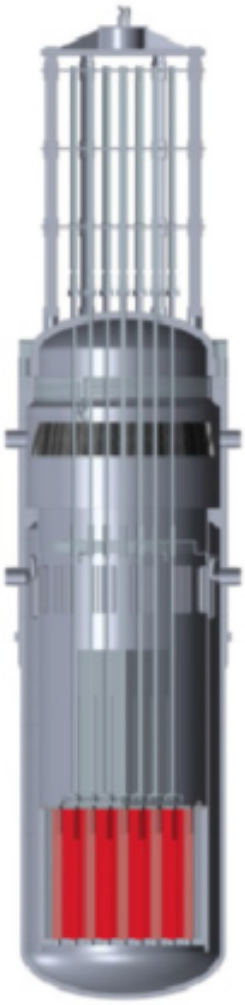


Figure 1: Reactor System Configuration of KARAT-100
(Reproduced courtesy of NIKIET)

2. Target Application

The KARAT-100 is a multipurpose BWR assigned for electricity generation, district heating and for cogeneration. The KARAT-100 power unit has a high load following capability to cope with daily power variation from 20% to 100% of nominal capacity.

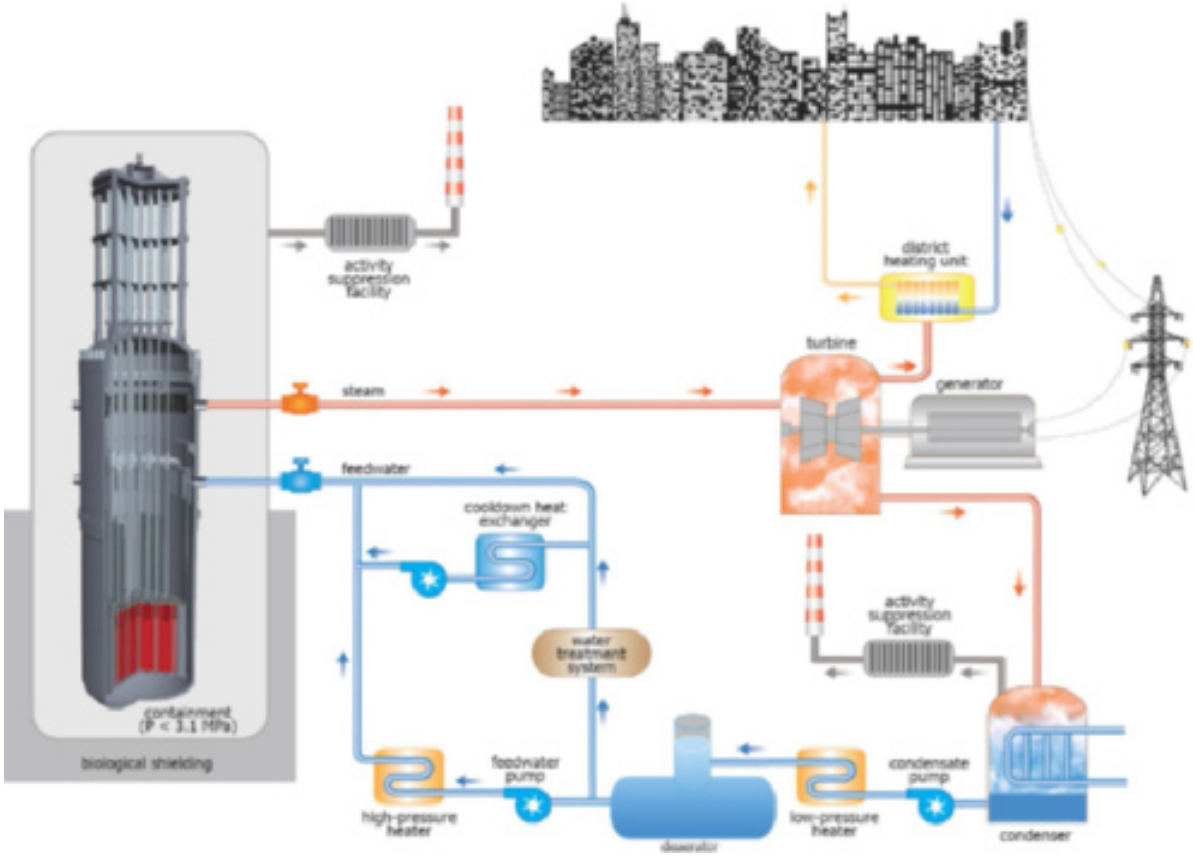


Figure 2: The KARAT-100 multipurpose application
(Reproduced courtesy of NIKIET)

3. Development Milestones

2014	Development of Technical Proposal
2016	Conceptual design
2017-2019	Basic design, R&D
2020	Start of detailed design and equipment fabrication
2023	Commissioning

4. General Design Description

Design Philosophy

KARAT-100 reactor is being built as the base reactor for the evolution of power generation in isolated or remote locations not connected to the unified grid. The key factor that makes this reactor a perfect choice for a nuclear cogeneration plant is its economic competitiveness against other sources of thermal and electric power, achieved primarily due to a combined generation of heat (for district heating) and electricity.

Nuclear Steam Supply System

The reactor uses a single-circuit heat removal system. Steam is generated by the coolant boiling in the reactor core. The steam discharge system is designed to remove steam from the reactor directly to the turbine plant. The steam pressure in the system is 7 MPa, at the steam temperature of 286 C. The humidity of the steam fed to the turbine plant is 0.1%. KARAT-100 uses gravity-type steam separation with additional drying in louver-type separators without centrifugal axial separators, which improves the stability of the reactor operation

Reactor Core

The reactor core as shown in *Figure 3*, consists of 199 FAs of proven design. The total number of cells in the core support grid is 253 (there are locations for an extra FA row or the reflector). There are two types of FAs used in the reactor core: 111 with channels for the CPS rods and 88 without such channels.

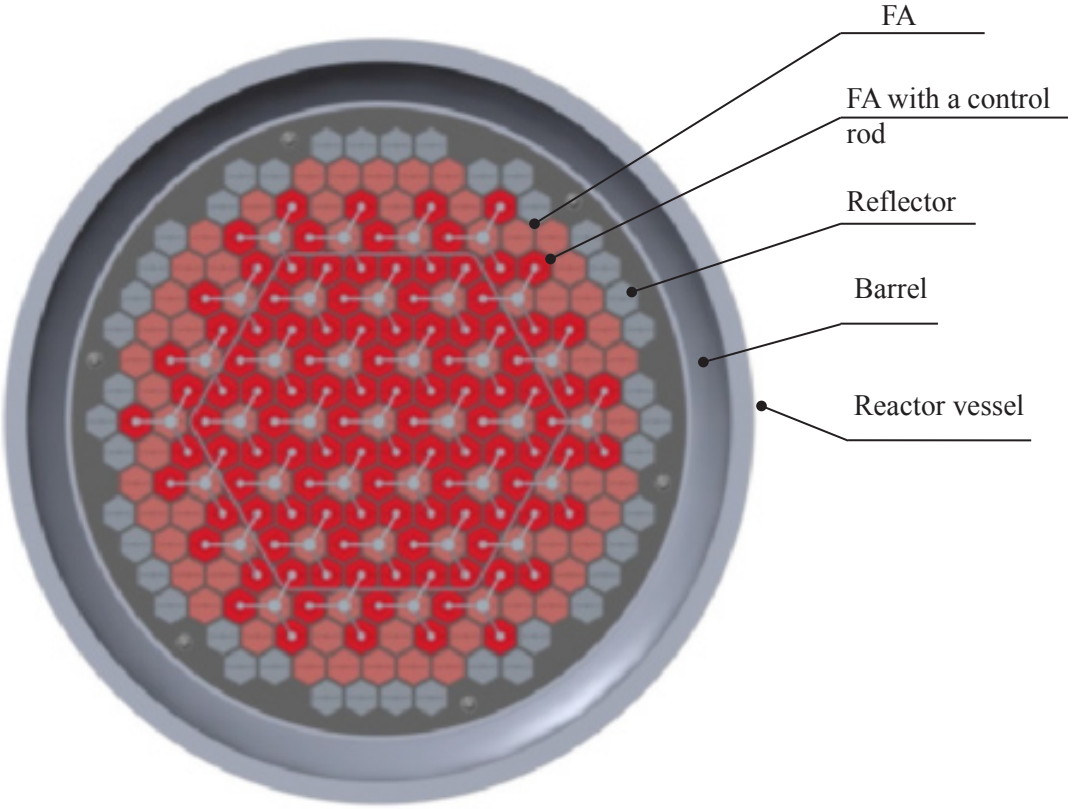


Figure 3: KARAT-100 reactor core
(Reproduced courtesy of NIKIET)

Reactivity Control

The reactor core includes 111 control and protection system (CPS) rods. The rod represents a structure comprising eight absorber elements arranged uniformly in a circumferential direction and retained in a support grid. The absorber elements are spaced by a spacer grid. The absorber material is boron carbide (B_4C). The rods are accommodated in the FA guide channels and are grouped into clusters of 3 assemblies each to reduce the number of actuators.

Reactor Pressure Vessel and Internals

KARAT-100 reactor vessel consists of a number of shells, a head and a bottom welded one to another. The reactor internals - a barrel, a core support grid, a chimney, emergency heat exchangers and the reactor core - are accommodated inside the vessel. The reactor vessel has nozzles for feed water supply and steam discharge, as well as nozzles for the emergency heat exchangers. All nozzles are located in the vessel's upper part which guarantees that the necessary volume of coolant is maintained even in the event of a nozzle break.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	NIKIET, Research and Development Institute of Power Engineering
Country of origin	Russian Federation
Reactor type	Boiling Water Reactor
Electrical capacity (MW(e))	100
Thermal capacity (MW(th))	360
Expected capacity factor (%)	92
Design life (years)	80
Plant footprint (m^2)	22500
Coolant/moderator	Light water
Primary circulation	Natural circulation
System pressure (MPa)	7.0
Core inlet/exit temperatures (^oC)	104 / 286
Main reactivity control mechanism	Control Rod Drive Mechanism (CRDM)
RPV height (m)	13.25
RPV diameter (m)	4.00
RPV or module weight (metric ton)	350
Configuration of reactor coolant system	Typical natural circulation BWR
Power conversion process	Direct Rankine Cycle
Fuel type/assembly array	UO_2 pellet/hexagonal
Fuel assembly active length (m)	1.98
Number of fuel assemblies	199
Fuel enrichment (%)	4
Fuel burnup (MWd/kg U)	45.9
Fuel cycle (years)	7.5
Cogeneration capability	Yes
Approach to engineered safety systems	Passive
Number of safety trains	2
Refuelling outage (days)	30 (once in three years)
Distinguishing features	Multi-purpose reactor: electricity generation, heat production and nuclear cogeneration plant
Modules per plant	1 - 4
Target construction duration (months)	48
Seismic design	3g
Core damage frequency (per reactor year)	1E–6
Design Status	Conceptual design

Reactor Coolant System

The coolant is desalinated light water. The heat removal system is single-circuit. Steam is generated directly in the reactor vessel and, after being dried in the steam space, is fed to the turbine with a humidity of not more than 0.1 wt.% using gravity separation and louver-type separators built in the reactor. The steam is dumped into the condenser downstream of the turbine. Some part of the steam could be extracted for heating of plant system's in house water and for heat supply.

Pressurizer
There is no pressurizer. Pressurization is achieved through negative feedbacks on temperature, power and void reactivity effect.

5. Safety Features

The major goal of the safety assurance arrangements is to limit the KARAT-100 radiological impacts on the personnel, the public and the environment during normal operation and in cases of operational occurrences and emergency events. KARAT-100 safety is ensured through the technological sophistication of design, the required fabrication, installation, adjustment and testing quality and robustness of the reactor facility's safety related systems and components, operating condition diagnostics, quality and timeliness of the equipment maintenance and repair, in-service monitoring and control of processes, organization of work, and qualification and discipline of personnel.

Engineered Safety System Approach and Configuration
KARAT-100's system of engineered and organizational measures forms five defence-in-depth levels:

- Conditions for KARAT-100 siting and prevention of anticipated operational occurrences;
- Prevention of design-basis accidents by normal operation systems;
- Prevention of beyond design-basis accidents by safety systems;
- Management of beyond design-basis accidents;
- Emergency planning.

Decay heat removal system
Residual heat is removed during an accident with a loss of heat removal by normal operation systems with the help of coil-type emergency heat exchangers accommodated inside the reactor vessel and emergency cooling tanks. The system is based on a passive principle of action. The coolant is discharged as steam depending on the decay heat level.

Emergency core cooling system
The reactor is cooled down in emergencies caused by a loss of the primary circuit integrity or the reactor power supply using six independent channels through:

- The generator coast down;
- The passive decay heat removal system, including the emergency heat exchanger;
- The emergency cooldown system;
- Passive-type water accumulators;
- The boron solution injection system;
- Cooling the reactor's metal containment.

Additionally, the power unit with KARAT-100 reactor is equipped with the following safety systems:

- A steam localization system downstream of safety valves required to localize radioactive steam release when the safety valves actuate;
- The system for the return of boron solution into the reactor designed to feed the borated coolant back from the reactor cavity in the event of a reactor vessel or nozzle break;

- The reactor water accumulation system aimed at keeping the water inventory in the accumulators for making up the reactor in emergencies caused by a decrease in the reactor vessel coolant level;
- The emergency power supply system used in the event of a loss of power supply from the energy grid.

Containment system

KARAT-100 reactor is housed in reinforced-concrete containment with a stainless steel liner. The containment forms an additional barrier against the release of radioactive substances into the environment while limiting at the same time, by its volume, the coolant loss in the event of reactor vessel break. There are isolation gate valves installed on pipelines at the containment outlet.

6. Plant safety and Operational Performances

The major goal of the safety assurance arrangements is to limit KARAT-100 radioactive impacts on the personnel, the public and the environment during normal operation, and in cases of operational occurrences and emergency events. KARAT-100 safety is ensured through the specific transfer and distribution of radioactive substances due to water boiling. The key factors are:

- A high inter-phase barrier (water-steam) to the spreading of nongaseous radionuclides prevents these from entering the steam-condensate line;
- Continuous degassing of coolant and removal of gaseous fission products from the circuit limit their accumulation in the circuit.

7. Instrumentation and Control systems

In-core monitoring allows thermal-hydraulic and neutronic parameters of the reactor core and the in-core coolant natural circulation circuit to be measured directly and indirectly in different operating modes of the reactor. KARAT-100 reactor is expected to be monitored for neutronic and thermal parameters, including the reactor water level, the core inlet water temperature, and the steam space temperature, and periodically tested for the water chemical properties.

8. Plant Arrangement

The building layout plan for the land-based power unit of KARAT-100 is designed in such a way that the reactor system, including its servicing systems, spent fuel pool, and auxiliaries are located in double protective air-crash resistance buildings. The designers also claim that the overall size of the steam generating unit allows transportation of the reactor by railway.

Reactor building

The reactor unit building performs the function of a primary containment. The reactor unit houses the KARAT-100 reactor as well as the systems responsible for the emergency removal of heat from the reactor, the emergency reactor shutdown and the removal of radiolysis products from beneath the reactor head. Besides, the reactor building houses an irradiated FA storage facility and its cooling system, as well as the reactor facility's handling equipment.

Figure 4: Plant Layout arrangement of the KARAT-100
(Reproduced courtesy of NIKIET)

Control building
The main control room and emergency control room are located in control building adjoining the reactor unit building, from where the reactor facility is operated and thermal parameters are monitored.

Balance of plant:

Turbine Generator building
The turbine block houses the turbine generator, the steam condensate line components and equipment, and a bridge crane for moving operations. The dimensions of the turbine block are 42x28 m, and its height is 28.4 m.

Electric power systems
The normal power supply system is designed to supply electric power to all plant consumers during normal operation and anticipated operational snags, including accidents, as well as to deliver the electricity generated by the turbine plant to offsite and in-house consumers.

ELENA (Kurchatov Institute, Russian Federation)

1. Introduction

The ELENA nuclear thermoelectric plant (NTEP) is a direct conversion water-cooled reactor without on-site refuelling capable to supply 68 kW(e) of electricity and 3.3 MW(th) of heating capacity for 25 year without refuelling. The technology and techniques were developed incorporating the experience from the construction and operation of the GAMMA reactor for marine and space application. The ELENA NTEP is designed as an "unattended" nuclear power plant (NPP), requiring nearly no operating or maintenance personnel over the lifetime of the unit. The conceptual design 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 also possible to develop the underground or underwater versions. The reactor and its main systems are assembled from factory-fabricated finished components or modules, whose weight and dimensions enable any transport delivery for the complete plant, including helicopter and ship. The specific features of the design include:

- 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.

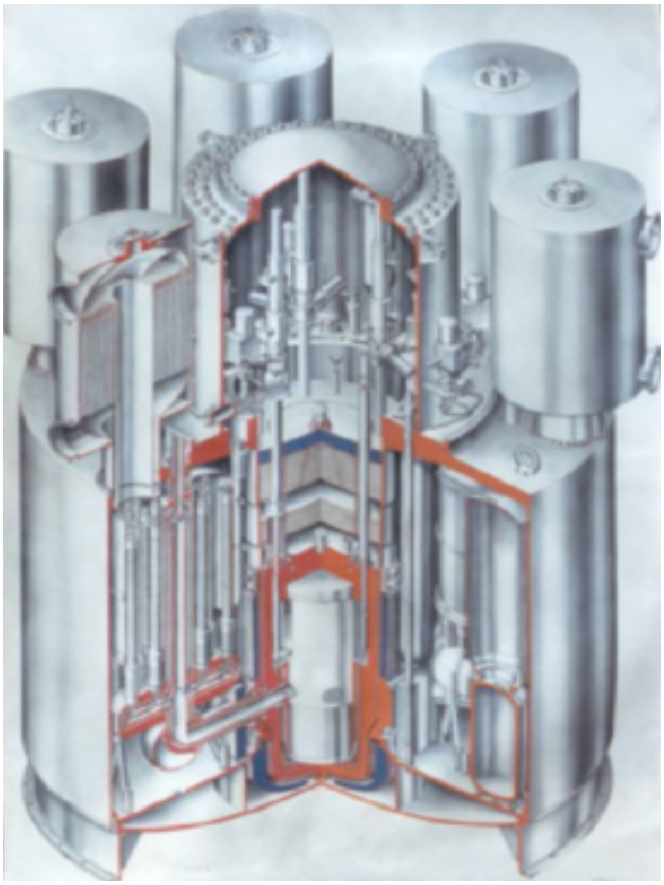


Figure 1: Reactor system configuration of ELENA (Reproduced courtesy of the Kurchatov Institute)

2. Target Application

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.

3. General Design Description

Design Philosophy

The ELENA reactor is designed with an integrated primary circuit. The design features of ELENA ensure high reliability and safety, eliminate adverse environmental impacts, and make the ELENA NPP an attractive source of heat and power supply for small settlements located in remote areas, including seismic and draught ones, as well as in uninhabited or underwater stations, e.g., robotized systems for investigation and extraction of ocean resources or hydrology research laboratories.

Nuclear Steam Supply System

The nuclear steam supply system (NSSS) consists of a reactor core and internals and the steam generators. The design is based on an integral reactor located in the large volume of secondary water. The NSSS is enclosed in a cylindrical vessel that is embedded in the reactor pool structure which is filled with water. Electric power is generated in semiconductor thermal battery due to the temperature difference provided between primary and secondary circuits.

Reactor Core

Pellet type uranium dioxide fuel is used with the average U_{235} 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 U_{235} load is 147 kg.

Reactivity Control

A reliable operation and reactivity control is achieved through the implementation of passive reactivity regulation and control systems. The control and safety systems, including the control rods and control rod drive mechanisms are used for reactivity control. The control and safety systems are designed to be fail safe. The ELENA reactor target to provide the total reactivity margin in a hot core as small as to secure the survival of an unprotected transient overpower with no core damage. It also ensures reactivity self-regulation throughout a very long period of unattended operation.

Reactor pressure vessel and internals

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.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology Developer	Russian Research Centre "Kurchatov Institute" (RRC KI)
Country of origin	Russian Federation
Reactor type	PWR
Electrical capacity (kW(e))	68
Thermal capacity (kW(th))	3300
Expected Capacity Factor	> 87
Design Life (years)	–
Plant Footprint (m^2)	–
Coolant/moderator	Light water
Primary circulation	Natural circulation
System pressure (MPa)	19.6
Core inlet/exit temperatures (^oC)	311/328
Main Reactivity Control Mechanism	Control rod drive and absorber rods
RPV Height (m)	3.7
RPV Diameter (m)	1.25
Weight (without primary and secondary coolant) (ton)	160
Configuration of Reactor Coolant System	Integrated
Power Conversion Process	Heating reactor, direct, thermo-electric
Fuel Type/Assembly Array	UO_2 pellets; MOX fuel is an option
Fuel Assembly Active Length (m)	–
Number of Fuel Assemblies	109
Fuel Enrichment (%)	15.2
Fuel Burnup (GWd/ton)	57600/27390
Fuel Cycle (months)	300
Cogeneration Capability	Possible
Approach to Engineered Safety Systems	Passive
Number of safety trains	–
Refueling Outage (days)	–
Distinguishing features	Twenty five (25) years life of the plant without refuelling, passive reactivity regulation and control systems and a very long period of unattended operation.
Modules per Plant	1
Target Construction Duration (months):	48 (including manufacturing of components)
Seismic design	–
Predicted core damage frequency (per reactor year)	–
Design Status	Conceptual design

Reactor coolant system

The ELENA reactor is a naturally circulated primary system with an integrated reactor coolant system. The complete reactor system is fabricated from stainless steel. Natural circulation of coolant in both circuits ensures the NPP unattended operation without on-site refuelling for up to 25 years. The temperature of the water within the third loop is ~100°C. The power level is primarily dependent upon the temperature of the third loop. The internal space for heat transport to consumers is connected to an air-cooled heat exchanger enclosed

in the draft tube for excess heat discharge to the atmosphere.

Primary Cooling Mechanism

The ELENA has three water coolant loops. The primary coolant loop is completely contained within the secondary barrier. Heat is transported from the core to the consumer though a four circuit system:

- The primary circuit (circuit I) with natural circulation of the coolant (water with a pressure of 19.6 MPa) transports heat from the core to the thermoelectric generator (TEG) modules cooled by the circuit II coolant (water with a pressure of 0.36 MPa);
- 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 is designed as a thermo-siphon with water or low-boiling coolant. Circuit III transfers heat through natural circulation to the heat exchanger of the heat supply circuit, the coolant being 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.

4. Safety Features

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 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.

Engineered Safety System Approach and Configuration

ELENA systems are designed with inherent safety features to ensure it remains in a safe configuration under any condition. The incorporation of defence-in-depth approach based on six safety barriers prevents the primary circuit from depressurization and secure activity confinement inside the reactor during accidents. The use of a self-adjustable water-cooled reactor coupled with thermoelectric mode of heat conversion and natural circulation of coolant makes it possible to exclude movable elements from the technological circuit of an NPP and to secure a lifetime unattended operation without on-site refuelling. 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. The active components of the protection system are scram actuators for six compensation groups of the control rods.

Decay heat removal system
A low specific thermal power of the ELENA reactor enables easy removal of the residual heat after the reactor shutdown. The residual heat is damped naturally to the compartment and the fuel elements are not super-heated during this process.

Emergency core cooling system
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.

Containment system
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. In turn, the caisson is encircled by the external containment, which is the next barrier to the spread of radioactivity; water that fills the containment volume is the circuit II coolant and acts as a biological shielding for the reactor. The external containment forms the cylindrical geometry of the plant with a height of 13 m and a diameter of 4.45 m.

5. Plant safety and Operational Performances

The ELENA reactor does not require an operator during nominal power operation of the plant. Operators are required for assembly, startup and to begin nominal operation. The reactor is designed to operate in a base load mode. 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. 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. There are no valves or mechanical parts which require maintenance over the lifetime of the plant. Once operational the ELENA reactor depends upon natural processes to maintain the reactor power without the actuation of control rods. The control and safety systems, including the control rods, control rod drive mechanisms and sensors are used only for the reactor startup, or for the times that the reactor is scrammed. Startup is done by an on-site operator who can leave the site once steady-state power has been obtained. The reactor startup is done by measuring the neutron flux and calculating the reactor period. The reactor outlet temperature and pressure in the coolant loop is monitored, but do not provide feedback through the control loop during startup. To begin the operation, the poison rods are pulled completely from the core, and are never inserted during nominal operation. To start up and reliably shut down the reactor in any situation, a grid is included that compensates the excessive reactivity. The compensation grid consists of six groups of the boron carbide absorber rods in stainless steel claddings of 1.45 cm external diameter. Each group (34 rods) has an individual drive.

6. Instrumentation and Control systems

The instrumentation and control (I&C) system of the ELENA reactor is specially arrangement to register parameter deviations at early stages of the accidental conditions to predict their further progression.

7. Plant Arrangement

The reactor system can be broken into two parts for shipment. It is possible to fuel the system

on-site, thereby eliminating problems associated with shipping a fuelled reactor. 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.

Reactor building

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 vessel head of the system is removable. The plant incorporates a physical protection system, has a fence and is equipped with external lighting.

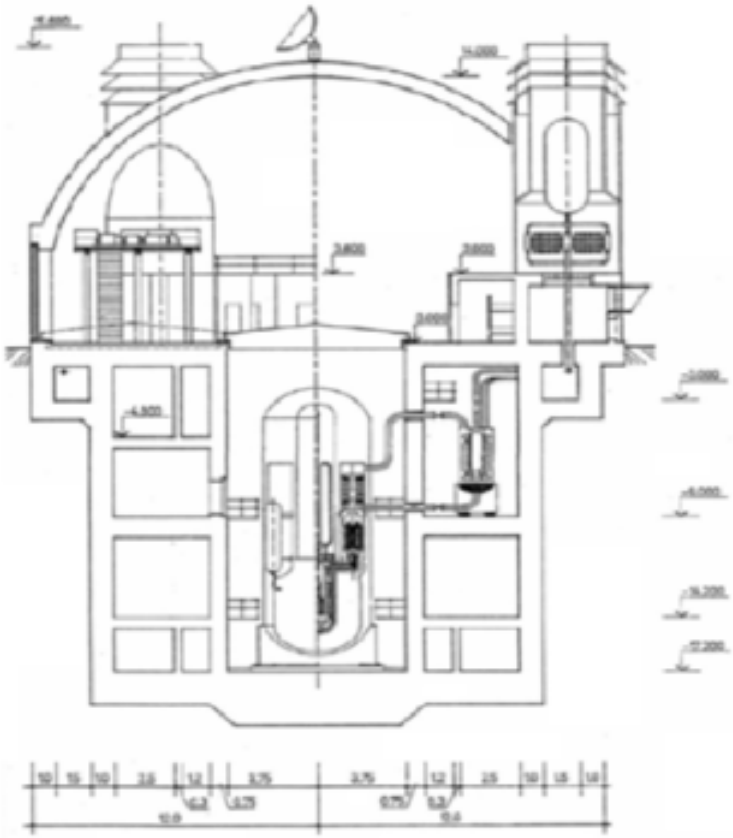


Figure 2: Plant general view of ELENA
(Reproduced courtesy of the Kurchatov Institute)

Control building

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.

***Balance of Plant*:**

Turbine Generator building

A 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.

Electric power systems

The ELENA-NTEP CSS has three independent power supply system: i.e., two (2) TEG sections, a diesel generator, and a storage battery. The electric power output can be controlled either by the use of shut resisters or by short circuiting the TEs. The TE power conversion system has low electrical conversion efficiency, and the waste heat is used for district heating.

8. Design and Licensing Status

The assembly drawings of the ELENA have been completed, and are ready for fabrication and testing of the system.

9. Plant Economics

The target cost of electricity in US $ is 0.06/kW-hour. The cost of energy for district heating is expected to be US$ 50/GCal. The ELENA NTEP is designed for long-term operation with a single fuel load and absence of the personnel, thus the labour payment and life support costs reduce considerably. Although the specific construction cost of the ELENA NTEP is projected to be higher, the power generation cost is expected to be competitive with the organic fuel for remote areas or for specific applications.

RUTA-70 (NIKIET, Russian Federation)

1. Introduction

RUTA-70 is a water-cooled water-moderated integral pool-type heating reactor with a thermal capacity of 70 MW(th). RUTA-70 is a dedicated reactor for district heating or potable water production; it has no energy conversion system. The reactor pool is made of reinforced concrete lined by stainless steel. In the primary cooling circuit, the heat from the core is transferred to the primary heat exchanger (HX) by forced convection at full power but by natural convection for powers below 30% of the rated power. The application of forced coolant circulation using pumps for operations at power levels of 30% to 100% rated power make it possible to increase the coolant flow rate in the primary circuit and raise the down-comer temperature by reducing the water thermal gradient in the reactor core. The reactor power is regulated using the control and protection system (CPS) drives, which are installed in the reactor pool water. The period of continuous operation of the reactor equipment without any 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 safety features, nuclear district heating plant (NDHP) using RUTA reactors can be located in the immediate vicinity of the heat users. Specific features of the RUTA are as follows:

- No high coolant pressure in the primary circuit (atmospheric pressure above the water level in the reactor pool);
- Large water inventory in reactor pool enables heat accumulating capacity;
- Low core power density (< 30 MW(th)/m^3);
- Natural convection is applied in core cooling for lower level below 30% of the rated power, as well as for cooldown modes;
- A three-circuit system of heat supply to consumers with the lowest water pressure kept in the primary circuit (primary pressure < secondary pressure < consumer circuit pressure).

2. Target Application

The conceptual design of RUTA is primarily developed to provide district heating in remotely isolated areas of the Russian Federation. Continuous increase of the organic fuel costs in the country essentially enhances the prospect of RUTA as a heating reactor. In addition, another potential thermal energy application of the RUTA is seawater and brackish water desalinations based on distillation process.

3. Design Milestones

1990	Conceptual design of the 20 MW(th) RUTA heating plant
1992	Feasibility study 'Design and Development of the underground nuclear heating plant 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 the 70 MW(th) RUTA reactor to upgrade the district heating system in Obninsk, Kaluga region.
	Feasibility study of constructing 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'

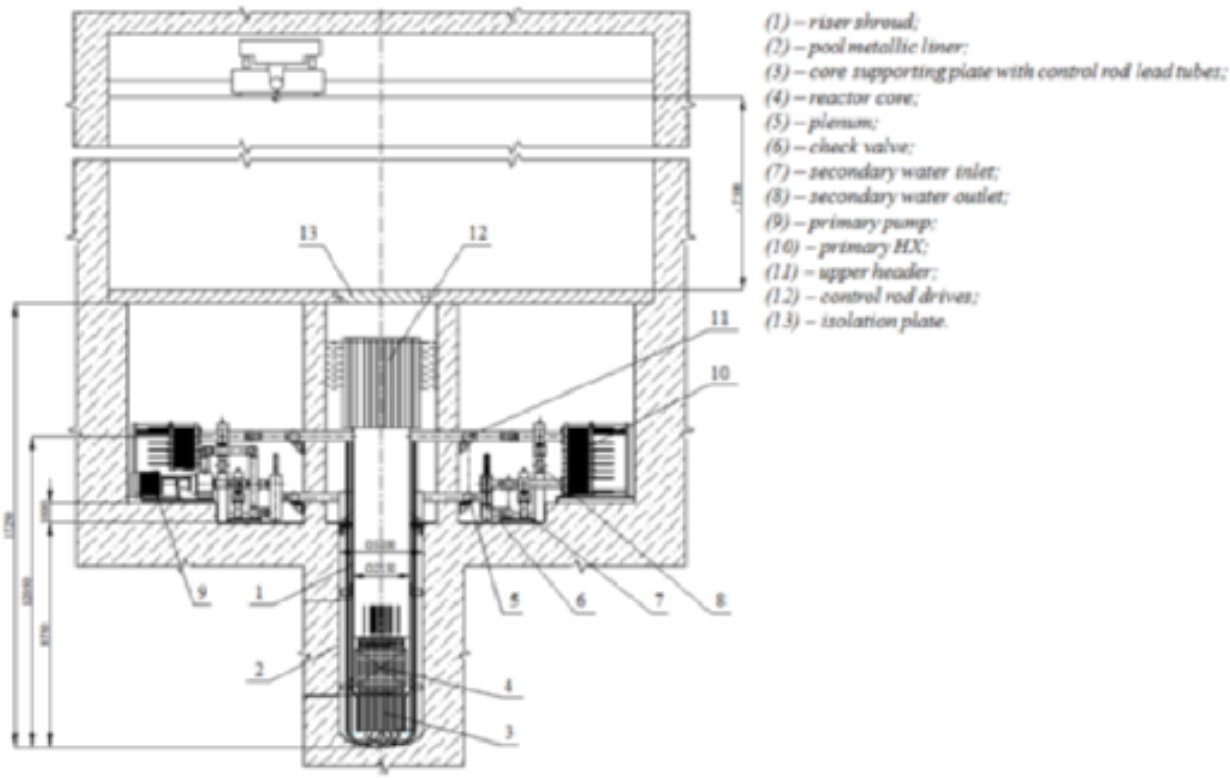


Figure 1: Reactor System Configuration of RUTA-70 (Reproduced courtesy of NIKIET)

4. General Design Description

Design Philosophy

The basic design principles of this reactor are simplicity of the design and a high safety level due to a low pressure and a large coolant inventory in the primary system. The concept of RUTA-70 reactor facility is based on simple design and, as a result, a low cost of plant construction and operation, and high level of safety achieved through the specific features of design and inherent safety mechanisms. The reactor facility is a part of the ground based nuclear heating plant (NHP) designed similarly to research reactors of the pool type.

Nuclear steam supply system

The RUTA reactor facility has a two-circuit layout. The primary circuit is an in-pile reactor core cooling circuit and the secondary circuit is an intermediate one that removes heat from the reactor and transfers it to the third circuit, which is the consumer circuit, i.e., to the heating network. Heat is transferred from the primary circuit to the secondary circuit and from the secondary circuit to the third circuit through the leak-tight heat-exchange surfaces, which are used to avoid the spread of radioactive products from the reactor circuit to the consumer. Most of the plant equipment, including the primary-to-secondary side heat exchangers (HX-1/2) resides at dry boxes outside the pool.

Reactor Core

The reactor core is placed in the lower part of the reactor vessel, the vault, in the shell of the chimney section. The core is designed with the 'Cermet' fuel rods that contribute to the reactor safety due to a high thermal conductivity of the fuel matrix and its role as the additional barrier to the fission products release.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	NIKIET
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	Light Water
Primary circulation	Natural (below 30% of N)/Forced (for 30-100% of N)
System pressure (MPa)	Atmospheric pressure at reactor poll water surface
Core inlet/exit temperatures (^oC)	75 / 102
Main reactivity control mechanism	Control rod driving mechanisms, Burnable poison
Reactor pool height (m)	17.250
Lower part of reactor pool diameter (m)	3.2
RPV weight (ton)	-
Configuration of reactor coolant system	Integrated pool-type
Power conversion process	N/A
Fuel type/assembly array	Cermet (0.6 UO_2+ 0.4 Al alloy) / hexagonal
Fuel assembly active length (m)	1.4
Number of fuel assemblies	91
Fuel enrichment (%)	3.0
Fuel burnup (GWd/ton)	25–30
Fuel cycle (months)	36
Cogeneration capability	Yes, for desalinations using distillation
Approach to engineered safety systems	Passive
Number of safety trains	2.0
Refuelling outage (days)	28
Distinguishing features	Designed for low temperature process heat, coupling with desalination system, radioisotopes production for medical and industrial purposes and creation of neutron beams for neutron therapy
Modules per plant	1.0
Target construction duration (months)	36
Seismic design (g)	>0.8 (automatic shutdown)
Predicted core damage frequency (per reactor year)	PSA is not completed, assessment < 1E–6
Design status	Conceptual design

The reactor core consists of 91 fuel assemblies (FA) of hexagonal geometry with 120 fuel rods per each FA. The height of reactor core is 1400 mm or 1530 mm depending on the fuel rod type. The core equivalent diameter is 1420 mm. In the radial direction, the design of the RUTA-70 fuel assembly is similar to that of the VVER-440 fuel assembly with the fuel assembly pitch of 147 mm and the fuel rod pitch of 12.2 mm. In the core, there are 42 reactor control and protection system (RCP) rods composing two shutdown systems with diverse actuators.

Reactivity Control
In the RUTA-70 design, the following mechanisms of reactivity control and power flattening are applied: optimization of refuelling, use of burnable poison, profiling of fuel loading and movable control rods. The reactivity control is performed by regulating the control rods and utilizing burnable poison. The reactivity margin is partly compensated by the burnable absorber (gadolinium) incorporated into the fuel rod matrix in a way to improve a core power distribution. The rest of the reactivity margin is compensated by the control rod groups. The control and protection system (CPS) controls the reactor power using the drives submerged into the reactor pool.

Reactor Coolant System
The primary coolant forced circulation is provided by two main circulation pumps - one pump per each of two reactor loops. Two MCPs of axial type are installed in the bypass lines of the main circulation loop close to the down-comer inlet. Pumps are located to allow easy access for inspection or replacement. The loop arrangement of the primary circuit components, with the secondary circuit pressure exceeding the pool water pressure, ensures that the reactor coolant is localised within the reactor tank.

Primary Cooling Mechanism
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 application of forced coolant circulation using pumps for operations at power levels of 30% to 100% rated power make it possible to increase the coolant flow rate in the primary circuit and raise the down-comer temperature by reducing the water thermal gradient in the reactor core. The distributing header is placed in the upper part of the shell of the chimney section. Pipelines of water supply to the primary HXs are connected to the header from both sides. HXs are located in the upper part of the pool. Under the natural convection operating mode, the forced convection circuit is isolated by a check valve installed at the pipeline of the individual section. Using the same valve, the natural convection circuit can be isolated. Downstream of the HXs, coolant is directed via the suction header to the circulation pump that supplies water to a group of heat exchangers located at one side of the pool. Water is returned from the pump head via the supply header. Pumps are connected to the bypass line of the natural convection circuit and are placed in a special compartment in close vicinity to the reactor pool.

5. Safety Features

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 safety level of pool reactors is achieved through their design features, which make it possible to resolve some of the major safety issues through the employment of the naturally inherent properties of the reactor. The safety concept of the RUTA-70 is based on the optimum use of inherent safety features, consistent implementation of defence in depth strategy and to perform the functions based on principles such as multi-channelling, redundancy, spatial and functional independence, application of a single failure criterion and diversity. 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.

Engineered Safety System Approach and Configuration

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 pre-stressed 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. There is a stabilisation of reactivity feedbacks determined by negative fuel and coolant temperature reactivity coefficients and by the positive density reactivity coefficient. This means that the heating up of the core structural components, including fuel, or the boiling of water in the core results in spontaneous reduction or self-limitation of the reactor's nuclear power at any initial position of the control rods, including the scram rods. 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. 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. 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.

Decay Heat Removal System

Natural circulation in the secondary circuit provides for residual heat removal from the shutdown reactor and passive cooldown of the reactor facility in blackout emergency situation. The passively actuated ASEC provides residual heat removal to the ultimate heat sink (atmospheric air). ASEC is envisaged for reactor cooldown in case of loss of auxiliary power. Each loop of the secondary circuit has an ASEC subsystem (train); the ASEC is connected at the bypass line of the network heat exchangers. 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 heat-up 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.

Emergency Core Cooling System

In emergency situations, residual heat is transferred by natural circulation of the coolant in the reactor tank and in the secondary circuit in station blackout condition. Heat is removed from the secondary circuit convectors using the ASEC under forced or natural circulation of air in the convector compartments. Direct-acting devices open air louvers of the ASEC passively. The system for emergency makeup of the primary and secondary circuits is an active system.

Reactor pool

The reactor pool consists of reactor core and internals, control and protection system, distributing and collecting headers and a large amount of water. Water in the reactor pool have a high heat accumulating capability that 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.

Containment system
The inner surfaces of the pool concrete walls are plated with stainless steel.

6. Plant Safety and Operational Performances

The NHP RUTA-70 may operate both in the base load and load follow modes. Two independent systems based on diverse drive mechanisms are provided for safe reactor shutdown and ensure the reactor power control. One system acts as an accident protection system, while the actuated second system is designed to provide guaranteed sub-criticality for an unlimited period of time and to be able to account for any reactivity effects including those in accidental states. Either system can operate under the failure of a minimum of one rod with maximum worth. In case of loss of power to the reactor control and protection system (RCP), all rods of this system can be inserted in the core under the effect of gravity.

7. Plant Arrangement

Reactor building
The protective flooring composed of slabs is 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.

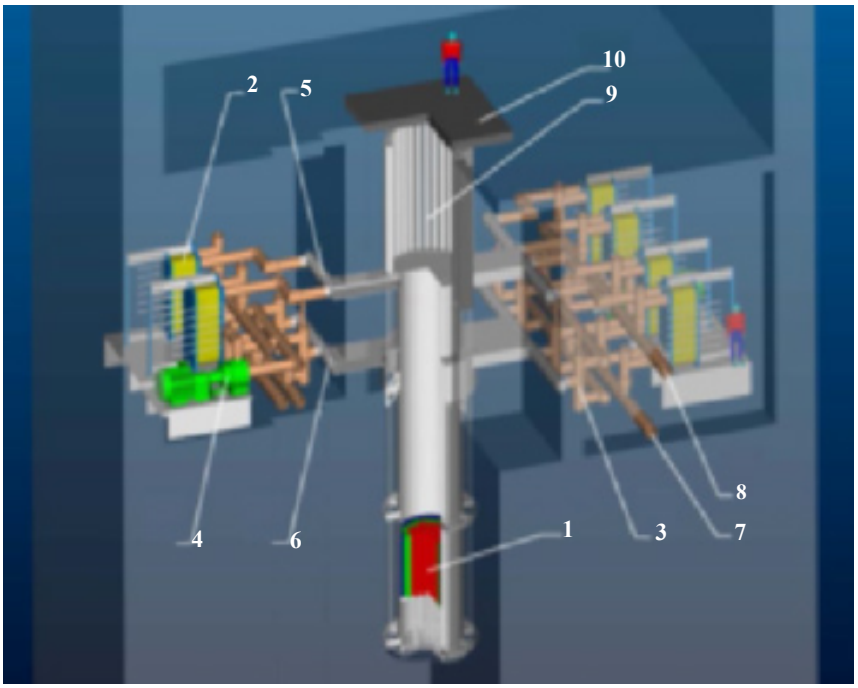


Figure 2: RUTA-70 Plant layout (Reproduced courtesy of NIKIET)

(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

Control building

The smallest staffing of the operating shift is four persons. These are the NDHP shift supervisor, the chief reactor control engineer, a fitter-walker for normal operation systems and a duty electrician to attend to electrical devices and systems, instrumentation and control. The respective services are responsible for radiation monitoring during operation and repair. A supervising physician and a refuelling operator are added to the regular shift staff for the core fuelling, first core critical mass attaining, power startup and refuelling periods. With managers, engineers and technicians working in one shift (a chemical engineer, an engineer for electrical devices and systems, an engineer for instrumentation and industrial control) and the administrative staff taken into account, the total personnel number is about 40 persons.

Balance of Plant:

The turbine and associated systems are not used in the NHP RUTA-70. RUTA-70 is a multi-purpose NHP for district heating, desalination and radioisotopes production for medical and industrial purposes. The principal scheme of NHP with RUTA reactor integration into the district heating system along with peak/backup boiler is show in *Figure 3* below. Such scheme in spite of some reduction of autonomy of district heating system (in comparison to high-temperature reactors) possesses several major advantages: Increase in reliability of heat supplying due to diversification of heat sources, provide redundancy required by relatively cheap heat sources and increase of economic effectiveness of heat production due to system optimization.

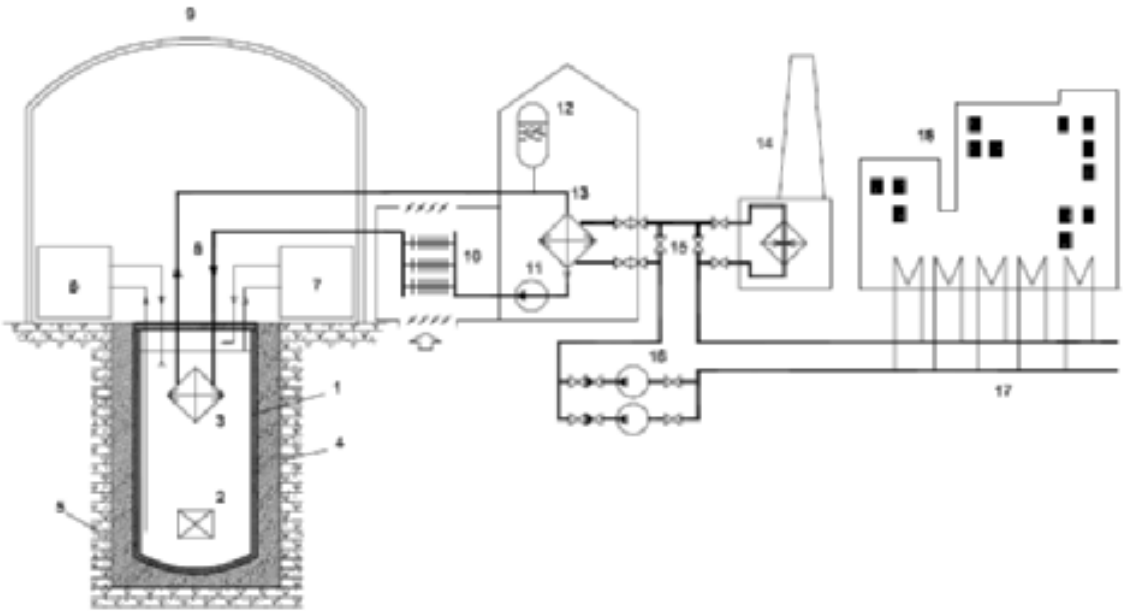


Figure 3: Scheme of NHP with RUTA reactor integrated in district heating system (Reproduced courtesy of NIKIET)

(1) reactor pool; (2) reactor core; (3) primary heat exchanger; (4) concrete vessel; (5) soil; (6) purification system; (7) ventilation system; (8) secondary circuit; (9) containment; (10) residual heat removal system; (11) secondary circuit circulation pump; (12) secondary circuit pressurizer; (13) secondary heat exchanger; (14) peak/backup heat source; (15) control valves; (16) grid circulation pumps; (17) grid water; (18) consumers.

8. Design and Licensing 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.

9. Plant Economics

The results of economic assessments show vast potential market for nuclear heating plant with RUTA-70 reactors. Duration of the NHP construction period is 3 years, including adaptation of the standard design to local site conditions. The projected specific capital cost in US $ is 279.6 per kW(th). Simplicity of design, low parameters, space-saving heat exchanging equipment and the use of passive systems contribute to the reduction of capital costs for construction of the NHP RUTA-70. The annual operating costs for the NHP RUTA-70 is 1772.5 US $ per year. The O&M costs can be reduced due to the limited number of required operating personnel and low costs for the repair and consumables. Further R&D is being performed for the reduction of operating costs by changeover to operating life of 100 years.

NuScale (NuScale Power Inc., United States of America)

1. Introduction

The NuScale Power Module (NPM) is a small, light water cooled pressurized-water reactor (PWR).The NuScale plant is scalable and can be built to accommodate a varying number of NPMs to meet a customer's energy demands. The 50 MW(e) NPM provides power in increments that can be scaled to 600 MW(e) gross in a single facility. A twelve module configuration is the current reference plant size for design and licensing activities. Each NPM is a self-contained module that operates independently of the other modules in a multi-module configuration, as shown in *Figure 1*. All modules are managed from a single control room. The plant design contains following significant features:

- reliable and passively safe systems that are simple in design and operation;
- safety features that assure a core damage frequency substantially lower than the current light water reactor fleet;
- reactor building designed to protect against aircraft impact while maintaining spent fuel pool integrity and preserving containment integrity;
- 60-year design plant life;
- reduced construction schedule compared to nuclear plants of a comparable output;
- modularization to enable in-shop fabrication of reactor and containment components; and
- use of half-height traditional 17x17 nuclear fuel assemblies.

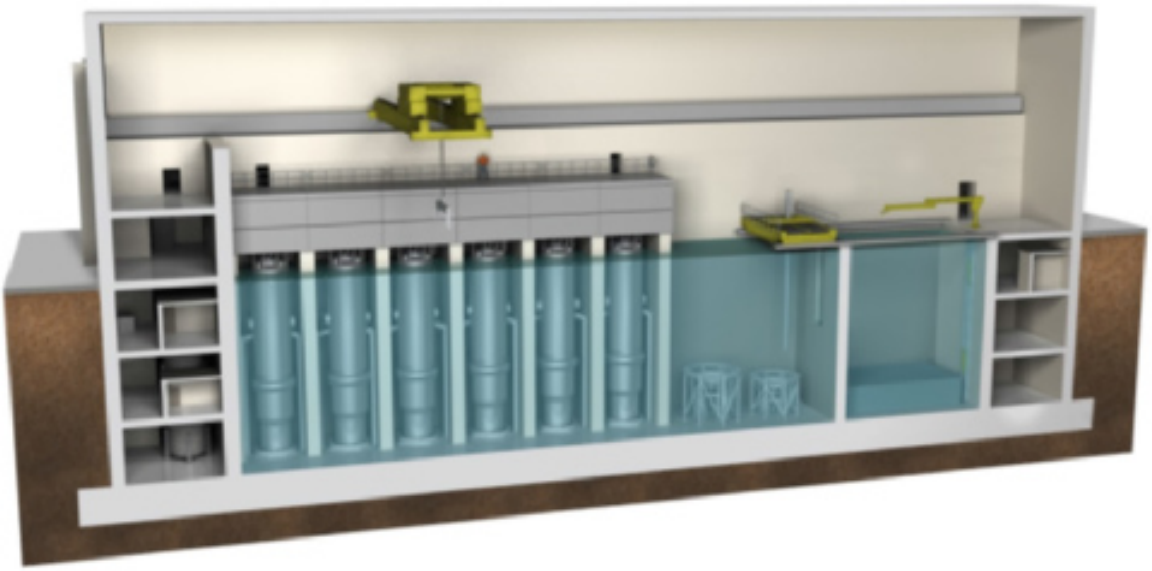


Figure 1: Cut-away view of NuScale power plant
(Reproduced courtesy of NuScale Power Inc.)

2. Target Applications

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

3. 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 (2nd half)	NuScale to submit design certification application to the NRC
2017 (2nd half)	First customer to submit application for a combined construction and operation licence (COL)
2020	NuScale design certification target

4. General Design Description

Design Philosophy

The NuScale plant design philosophy consists of: a design that uses proven light-water reactor technology, modular nuclear steam supply system, factory-fabricated power modules with shippable components, and passive safety systems that allow for unlimited coping time after a design basis accident without power, operator action, or makeup water. The NPM is designed to operate efficiently at full-power conditions using natural circulation as the means of providing core coolant flow, eliminating the need for reactor coolant pumps.

Nuclear Steam Supply System

The nuclear steam supply system (NSSS) consists of a reactor core, helical coil steam generators, and a pressurizer within a reactor pressure vessel (RPV). The NSSS is enclosed in an approximately cylindrical containment vessel (CNV) that sits in the reactor pool structure as shown in *Figure 2*. The reactor core is located below the helical coil steam generators inside the RPV as shown in *Figure 3*.

Figure 2: Nuclear Steam Supply System

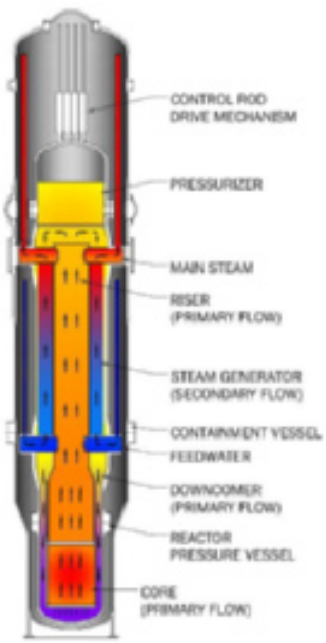


Figure 3: Diagram of the NuScale Power Module

(Reproduced courtesy of NuScale Power Inc.)

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer	NuScale Power, LLC
Country of origin	USA
Reactor type	Integral PWR
Electrical capacity (MW(e))	50 (gross)
Thermal capacity (MW(th))	160
Expected Capacity Factor	>95%
Design Life (years)	60
Plant Footprint (m^2)	130000
Coolant/moderator	Light water
Primary circulation	Natural circulation
System pressure	12.8 MPa
Core inlet/exit temperatures (^oC)	-- / 302
Main Reactivity Control Mechanism	Control rod drive, boron
RPV Height (m)	17.4
RPV Diameter (m)	2.9
Module Weight (metric ton)	--
Configuration of Reactor Coolant System	Integrated
Power Conversion Process	Indirect Rankine Cycle
Fuel Type/Assembly Array	UO_2 pellet/17x17 square
Fuel Assembly Active Length (m)	2
Number of Fuel Assemblies	37
Fuel Enrichment	<4.95%
Fuel Burnup (GWd/ton)	TBD
Fuel Cycle (months)	24
Cogeneration Capability	Possible
Approach to Engineered Safety Systems	Passive
Number of safety trains	2
Refueling 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
Target Construction Duration (months)	36
Seismic design	0.5g peak ground acceleration
Predicted core damage frequency (per reactor year)	1E-8 (internal events)
Design Status	Under development

Reactor Core

The core configuration for the NPM, as shown in *Figure 4*, consists of 37 fuel assemblies and 16 control rod assemblies. The fuel assembly design is modelled from a standard 17 x 17 PWR fuel assembly with 24 guide tube locations for control rod fingers and a central instrument tube. The assembly is nominally half the height of standard plant fuel and is supported by five spacer grids. The fuel is UO_2 with Gd_2O_3 as a burnable absorber

homogeneously mixed within the fuel for select rod locations. The U^{235} enrichment is below the current U.S. manufacturer limit of 4.95 percent enrichment. The reactor has a refuelling cycle of 24 months.

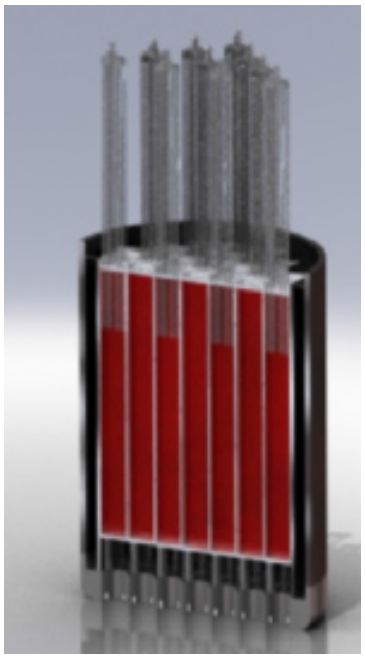


Figure 4: Reactor Core Assembly (Reproduced courtesy of NuScale Power Inc.)

Reactivity Control

Reactivity control in NPM is achieved mainly through soluble boron in the primary coolant and 16 control rod assemblies. The control rods are organized into two groups: a control group, and a shutdown group. The control group, consisting of four rods symmetrically located in the core, functions as a regulating group that is used during normal plant operation to control reactivity. The shutdown group comprising 12 rods is used during shutdown and scram events. Control rods absorber material is B_4C and the control rod length is 2 meters.

Reactor Pressure Vessel and Internals

The RPV consists of an approximately cylindrical steel vessel with an inside diameter of approximately 9 ft. and an overall height of approximately 58 ft. and is designed for an operating pressure of approximately 1850 psia. The upper and lower heads are torispherical and the lower portion of the vessel has flanges to provide access for refuelling. The RPV upper head supports the control rod drive mechanisms. Nozzles on the upper head provide connections for the reactor safety valves, the reactor vent valves, and the primary system piping.

Reactor Coolant System

NPM has natural convection-cooled core which does not require reactor coolant pumps. The NSSS and CNV are immersed in reactor pool which ensures passive long-term cooling and enhanced fission product retention. The reactor core is located inside a shroud connected to the hot leg riser. The reactor core heats reactor coolant causing the coolant to flow upward through the riser. When the heated reactor coolant exits the riser, it passes over the tubes of the helical coil steam generators, which act as a heat sink. As the reactor coolant passes over the steam generator tubes, it cools, increases in density, and naturally circulates down to the reactor core, where the cycle begins again.

Steam Generator
Each NPM uses two once-through helical-coil steam generators for steam production. The steam generators are located in the annular space between the hot leg riser and the RPV inside diameter wall. The steam generator consists of tubes connected to feed and steam plenums with tube sheets. Preheated feedwater enters the lower feed plenum through nozzles on the RPV. As feedwater flows through the interior of the steam generator tubes, heat is added from the reactor coolant. The secondary side fluid is heated, boiled, and superheated to produce steam for the turbine-generator unit.

Pressurizer
The pressurizer provides the primary means for controlling reactor coolant system pressure. It is designed to maintain a constant reactor coolant pressure during operation. Reactor coolant pressure is increased by applying power to a bank of heaters installed above the pressurizer baffle plate. Pressure in the reactor coolant system is reduced using spray provided by the chemical and volume control system (CVCS).

5. Safety Features

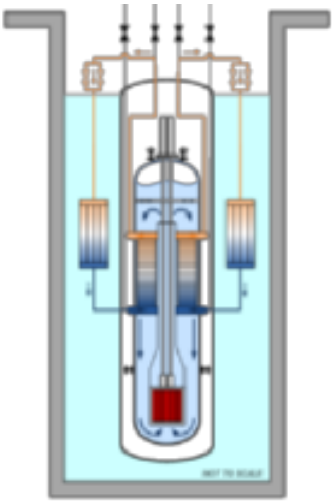
Engineered Safety System Configuration and Approach
Each NPM incorporates several simple, redundant, and independent safety features, which are discussed as follows:

Decay Heat Removal System
The decay heat removal system (DHRS) as shown in *Figure 5*, provides secondary side reactor cooling for non-LOCA events when normal feedwater is not available. The system is a closed-loop, two-phase natural circulation cooling system. Two trains of decay heat removal equipment are provided, one attached to each steam generator loop. Each train is capable of removing 100 percent of the decay heat load and cooling the reactor coolant system. Each train has a passive condenser immersed in the reactor pool. During normal operations, the condensers are maintained with sufficient water inventory for stable and effective operation.

Emergency Core Cooling System
Emergency core cooling system (ECCS) as shown in *Figure 6*, consists of two independent reactor vent valves (RVVs) and two independent reactor recirculation valves (RRVs). For LOCAs inside containment, the ECCS returns coolant from the CNV to the reactor vessel. This ensures that the core remains covered and that decay heat is removed. The ECCS provides a defense-in-depth means of decay heat removal in the unlikely event of a loss of feedwater flow, combined with the loss of both trains of the DHRS. The ECCS removes heat and limits containment pressure by steam condensation on, and convective heat transfer to, the inside surface of the CNV. Heat is then transferred by conduction through the CNV walls and convection to the water in the reactor pool. Long-term cooling is established via recirculation of reactor coolant to the RPV through the ECCS recirculation valves.

Reactor Pool
The reactor pool is a large stainless steel lined pool located below the plant ground level. Water in the pool provides cooling for the NPM for a minimum of 72 hours following any design basis accident. During normal plant operations, heat is removed from the pool through a cooling system and ultimately rejected into the atmosphere through a cooling tower or other external heat sink. In an accident where off-site power is lost, heat is removed from the NPMs by allowing the pool to heat up and boil. Water inventory in the reactor pool is large

enough to cool the NPMs for an unlimited time without adding water.



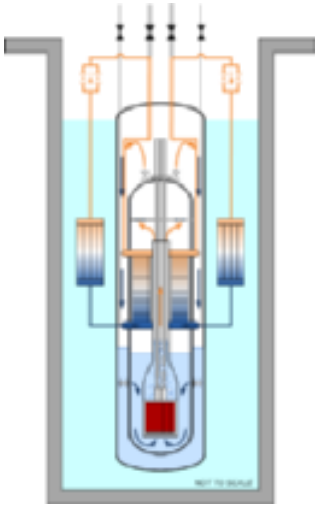


Figure 5: Decay Heat Removal System *Figure 6: Emergency Core Cooling System*
(Reproduced courtesy of NuScale Power Inc.)

Containment System

The major safety functions of the Containment Vessel (CNV) are to contain the release of radioactivity following postulated accidents, protect the RPV and its contents from external hazards, and to provide heat rejection to the reactor pool following ECCS actuation. Each CNV consists of a steel cylinder with an outside diameter of approximately 15 ft. and an overall height of approximately 76 ft. The CNV houses the RPV, control rod drive mechanisms, and associated piping and components of the NSSS. The CNV is immersed in the reactor pool, which provides a passive heat sink for containment heat removal under LOCA conditions. The CNV is designed to withstand the environment of the reactor pool as well as the high pressure and temperature of any design-basis accident. The CNV pressure is maintained at a vacuum under normal operating conditions. Maintaining a vacuum provides the thermal insulation of the RPV to parasitic heat loss during normal operation. The vacuum also reduces moisture inside containment that could contribute to component corrosion and impact the reliability of instrumentation and other systems within the CNV. Following an actuation of the ECCS, heat removal through the CNV rapidly reduces the containment pressure and temperature and maintains them at less than design conditions indefinitely. Steam is condensed on the inside surface of the CNV, and is passively cooled by conduction through the CNV and convection to the reactor pool water.

6. Plant Safety and Operational Performances

Each NPM is operated independent of other modules. A module is refuelled by disconnecting it from its operations bay and moving it to a common refuelling area within the shared reactor pool. The module is disassembled into to three major components: the lower RPV section that contains the core and lower internals, the lower CNV section, and the upper RPV/CNV section that contains the steam generators and pressurizer. After refuelling the core and inspecting the module sections, the module is reassembled and moved to its operations bay and reconnected to steam and feedwater lines. Other modules in the plant continue to operate while each module is refuelled.

7. Instrumentation and Controls Systems

The NuScale instrumentation and controls systems provide automated safety responses to specific initiating events such as reactor trip, followed by an integrated safety actuation of one or more of the passive safety systems; the DHRS and ECCS. The control room habitability system (CRHS) ensures that plant operators are adequately protected against the effects of accidental releases of toxic and radioactive gases. Areas served by the CRHS are maintained at positive pressure relative to adjacent areas. Compressed breathable air storage capacity can provide clean air to the control room spaces for at least 72 hours following an initiating event. The NuScale design effectively integrates human factors engineering (HFE) into the development, design, and evaluation of the plant. HFE employs state-of-the-art principles to ensure NuScale products facilitate safe, efficient, and reliable performance of operations, maintenance, tests, inspections, and surveillance tasks. The NuScale design minimizes human error through fail-safe design functionality, allows multi-modular control capability with effective automation design, employs digital display design and soft control technology to enhance usability, and provides optimum workload management.

8. Plant Arrangement

Reactor building

The NuScale plant consists of a power generation complex and common facilities. The power generation complex for the NuScale plant consists of up to 12 power generation units (12 NPMs and associated turbine generators), module assembly equipment, fuel handling equipment, turbine maintenance equipment, and radioactive waste processing equipment. A 12-unit plant net total output is approximately 570 MW(e). *Figure 7* presents the layout of a multi-unit plant. Each NPM is in its own bay immersed in the common reactor pool. Each reactor bay is approximately 20 ft. wide by 20 ft long and built into the nearly 75 ft. deep pool with a normal water level of approximately 69 ft. Each bay has a concrete cover that serves as a biological shield. The cover also serves to prevent deposition of foreign materials onto an NPM. The reactor pool is located in a Seismic Category I building designed to withstand postulated adverse natural conditions and aircraft impact.

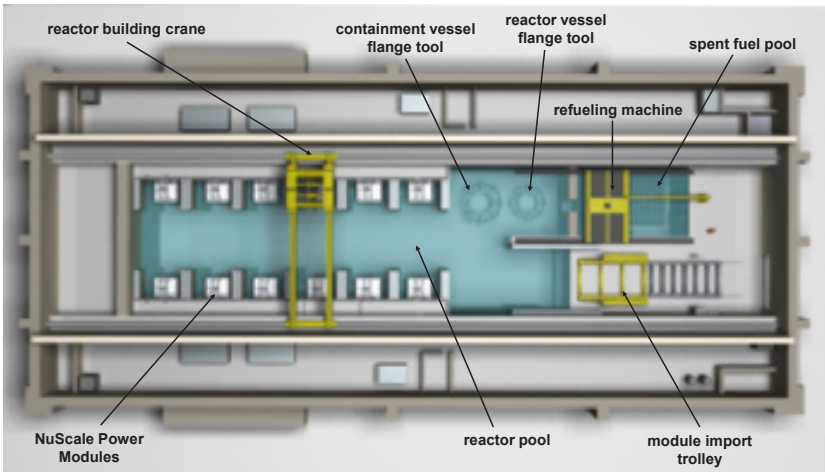


Figure 7: Reactor building plan view (Reproduced courtesy of NuScale Power Inc.)

Control building

The main control room is housed below grade in the control building located adjacent to the reactor building. The plant will have a control console in the main control room for all installed units. Each reactor operator will monitor and control multiple units from the control

room console. The reactor operators monitor the automated control system for each reactor. Each reactor is outfitted with monitors provided with soft controls and some select manual push buttons for operator control. The supervisor station provides an overview of all reactors using multiple monitors. All monitor displays are designed using human factors analysis to enhance simplicity. The display layout and design uses graphical representations of plant systems and components.

Balance of Plant:

Turbine Generator Building

A NuScale plant has two separate turbine buildings, each housing up to six turbine-generators. The turbine buildings are above-grade structures that house the turbine-generators with their auxiliaries, the condensers, condensate systems, and the feedwater systems. An emblematic turbine generator is shown in *Figure 8*. Each turbine-generator is associated with a single NPM and has dedicated condensate and feedwater pumps. Each turbine-generator is supported by an above-grade pedestal. Each condenser is located adjacent to the turbine-generator. An overhead crane is provided for the installation and maintenance in each turbine building. The turbine buildings are steel-framed structures with insulated metal wall siding and roof decking.

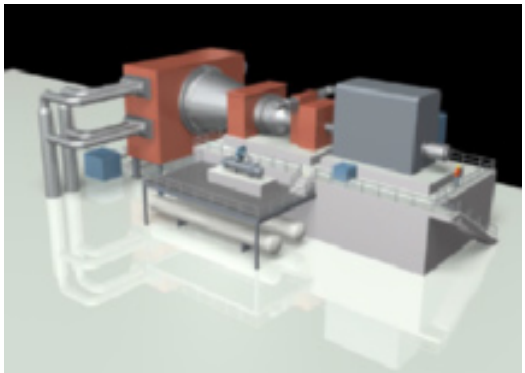


Figure 8: Turbine-generator (Reproduced courtesy of NuScale Power Inc.)

Electric Power Systems

Under normal operating conditions the AC electrical power distribution system supplies reliable and continuous power to equipment required for startup, normal operation, and shutdown of the plant. The NuScale plant does not require off-site AC electrical power to cope with design-basis events. No backup power is required for safety system actuation. In the event of failure of the AC electrical power supply, the DC backup supply system provides the necessary power to ensure continuous operation of post-accident monitoring instrumentation.

9. Design and Licensing 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 expected to be built in Idaho.

10. Plant Economics

The expected cost is less than $5000 per kilowatt for such a plant, amounting to a station cost of under $2.6 billion.

mPower (BWX Technologies, Inc., United States of America)

1. Introduction

The mPower™ plant consists of an integral PWR small modular reactor and related balance of plant, designed by Generation mPower LLC to generate a nominal output of 195 MW(e) per module. In its standard plant design, each mPower plant comprises a 'twin-pack' set, or two mPower reactor modules, generating a nominal 390 MW(e). The design adopts internal steam supply system components, once-through steam generators, pressurizer, in-vessel control rod drive mechanisms (CRDMs), and vertically mounted canned motor pumps for its primary cooling circuit and passive safety systems. The plant is composed of reactor modules that are fully shop-manufactured, rail-shippable to a site and installed into the facility. Additional 'twin pack' plants can be added on an as-needed basis to meet demand growth. The plant is designed to minimize emergency planning zone requirements. The mPower reactor adopts a number of features that aim to improve the constructability, reduce overall plant complexity and enhance availability. These include: proven structure, systems and components deployed commercial power plants; standard U^{235} fuel; modular rail and heavy truck shippable components; fewer, smaller and simpler NSSS component designs; advanced passive safety systems; and a two year refueling interval and conventional balance of plant systems and components.

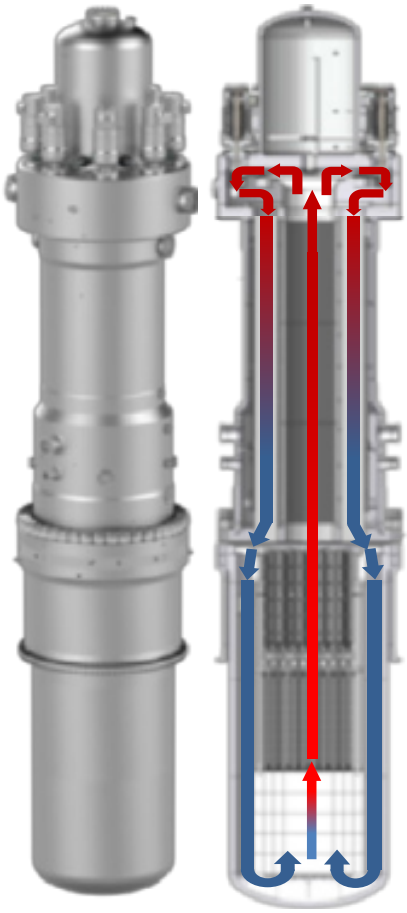


Figure 1: Reactor system configuration of mPower (Reproduced courtesy of Generation mPower LLC)

2. Target Application

The primary application for the mPower reactor is electricity production. The mPower design could be retrofitted to support other heat-requiring industries, desalination or co-generation applications.

3. Development Milestones

2009	BWX Technologies, Inc. (formerly B&W) officially introduced the mPower SMR concept
2010	Pre-application design certification activities engagement with the United States Nuclear Regulatory Commission
2012	The Integrated System Test (IST) facility located in Bedford County, Virginia, was put into operation
2014	Tennessee Valley Authority (TVA) announced its intention to submit an Early Site Permit Application at the TVA Clinch River site in Roane County for two or more SMR modules.
2015	The Babcock & Wilcox Company spun off its Power Generation business and the remaining company changed its name to BWX Technologies, Inc., retaining its interest in Generation mPower LLC and related nuclear steam supply system (NSSS) design authority
2016	BWXT and Bechtel Power Corporation agree to a Framework Agreement which provides for transition to a new management structure with Bechtel responsible for Program Management of the mPower program.

4. General Design Description

Design Philosophy

The mPower design is based on the use of systems and components with an advanced plant architecture that reduces licensing and construction risks. The mPower design employs passive safety features according to the defense-in-depth principle, including an underground steel containment vessel structure and an underground spent fuel storage pool.

Nuclear Steam Supply System

The NSSS consists of a reactor core, a steam generator (SG), reactor coolant pumps (RCPs), pressurizer and the core internals that are integrated within the reactor pressure vessel (RPV). The NSSS forging diameter allows greater sourcing options and rail shipments.

Reactor Core

The reactor core consists of 69 fuel assemblies (FAs) that have less than 5% enrichment, Gd_2O_3 spiked fuel rods, Ag In–Cd (AIC) control rods, and a design minimum 3% shutdown margin. The FAs are of a conventional 17×17 design with a fixed grid structural cage. FAs are shortened to an active length of 2.4 m and optimized to maximize fuel utilization. The operational cycle is 24 months with a fuel burn cycle of up to 48 months.

Reactivity Control

Soluble boron is eliminated from the reactor coolant. The primary means of reactivity control for the mPower design is achieved through the electro-mechanical actuation of control rods. The CRDM is fully submerged in the primary coolant within the RPV boundary which precludes the possibility of control rod ejection accident scenarios. Additional reactivity control is achieved through the use of the strong negative moderator temperature coefficient by control of the secondary side feedwater flow rates.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer	Generation mPower, LLC.
Country of origin	United States of America
Reactor type	Integral PWR
Electrical capacity (MW(e))	195
Thermal capacity (MW(th))	575
Expected Capacity Factor (%)	> 95
Design Life (years)	60
Plant Footprint (m^2)	157000
Coolant/moderator	Light water
Primary circulation	Forced circulation
System pressure (MPa)	14.8
Core inlet/exit temperatures (^oC)	290.5 / 318.9
Main Reactivity Control Mechanism	Control rods
RPV Height (m)	27.4
RPV Diameter (m)	4.15
Configuration of Reactor Coolant System	Integrated
Power Conversion Process	Indirect Rankine cycle
Fuel Type/Assembly Array	UO_2 pellet / 17x17 square
Fuel Assembly Active Length (m)	2.4
Number of Fuel Assemblies	69
Fuel Enrichment (%)	< 5
Fuel Burnup (GWd/ton)	> 40
Fuel Cycle (months)	24
Cogeneration Capability	Possible
Approach to Engineered Safety Systems	Passive
Number of safety trains	2
Refueling Outage (days)	< 25
Distinguishing features	Internal once-through steam generator, pressurizer and control rod drives
Modules per Plant	2
Target Construction Duration (months)	35
Seismic design	Target 85% of contiguous USA
Predicted core damage frequency (per reactor year)	Target 1E-8
Design Status	Under Development

Reactor Pressure Vessel and Internals

The mPower RPV houses the steam generator, CRDMs, pressurizer, reactor coolant pumps and the isolation valves. The integrated RPV inherently eliminates the possibility of a large break loss-of-coolant accident (LOCA). Reactor internals include core support, internal structures and all structural and mechanical elements inside the RPV.

Reactor Coolant System
The primary cooling mechanism of the mPower reactor under normal operating condition and shutdown condition is by forced circulation of coolant. The reactor uses eight reactor coolant pumps (RCPs) located on a 360° pump shelf at the top of the coolant riser. The large reactor coolant system (RCS) volume of the mPower reactor allows more time for safety systems to respond in the event of an accident. Additional cooling water is passively provided via the Emergency Core Cooling System (ECCS) for continuous cooling to protect the core during a small break LOCA.

Steam Generator
The steam generator (SG) is located within the annular space formed by the inner RPV walls and the riser surrounding and extending upward from the core. The upper vessel assembly including the SG is removed for access to the core during refueling and allows for inspection and maintenance in parallel with fuel exchange.

Pressurizer
The integrated electrically heated pressurizer located at the top of the RPV maintains a nominal 14.8 MPa. Reactor coolant pressure is controlled by the heaters and steam space spray.

5. Safety Features

The integral reactor is contained within a steel containment vessel located fully underground within the Reactor Service Building to provide enhanced protection against external events. The mPower plant safety features meet a seven-day coping time without off-site power.

Engineered Safety System Configuration and Approach
The inherent safety features of the reactor design include a low core linear heat rate that reduces fuel and cladding temperatures during accidents, a large RCS volume that allows more time for safety system responses in the event of an accident, and small penetrations at high elevations, increasing the amount of coolant available to mitigate a small break LOCA. The mPower plant deploys an enhanced spent fuel pool configuration, which is installed underground, with a large heat sink to cope up for 30 days in case of loss of fuel pool cooling.

Decay heat removal system
The mPower reactor deploys a decay heat removal strategy as shown in *Figure 2,* with an auxiliary steam condenser on the secondary system, water injection or cavity flooding using the refueling water storage tank, and passive containment cooling.

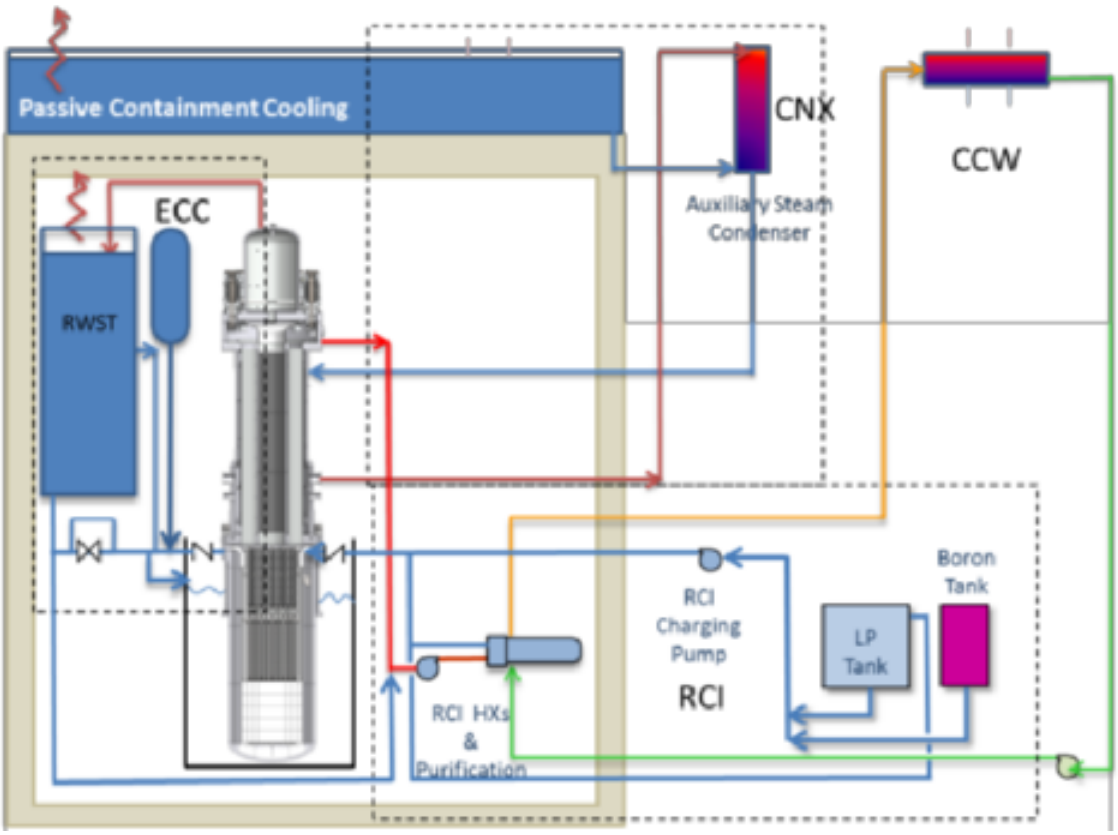


Figure 2: mPower plant decay heat removal strategy
(Reproduced courtesy of Generation mPower LLC)

Emergency core cooling system

The mPower ECCS is a safety system that provides three basic functions: (1) depressurization of the RCS, (2) reactor coolant inventory control during the event, and (3) core decay heat removal. With a system of automatic depressurization valves (ADVs) and the large coolant reserve provided by accumulators referred to as intermediate pressure injection tanks (IPIT) and the in-containment refueling water storage tank (RWST), the reactor core remains covered following a design-basis event. The IPITs are maintained with pressurized nitrogen over the water. Should the automatic depressurization valves open, the reactor pressure vessel will depressurize until in equilibrium with the containment atmosphere. During and following that pressure equalization, check valves between the reactor pressure vessel and IPITs (early) and the RWST (late) open, injecting gravity-driven cooling water from the RWST. The RWST provides a minimum of 7 days to as much as 14 days of cooling without the need for external intervention or AC power to maintain reactor core cooling and safe shutdown. To address the single-failure general design criteria, the ECCS is designed for n+1 components, with all components located inside containment. There are four high-pressure and four low pressure depressurization paths arranged in pairs connected to four pressurizer connections. The ADVs are the isolation valves for the eight lines. There are two injection paths to the RCS. Each path to the RCS is connected to an IPIT and to the RWST.

Containment system

The steel containment vessel and interfacing safety systems work in concert to protect the core, provide long-term core cooling, and prevent the release of radioactive materials to the environment without reliance on AC power or operator action well beyond the regulatory expectation of 72 hours following an accident. This seismic Category I structure is designed

to withstand the maximum internal pressure from design basis accidents, including LOCA and steam line break. Normal access to the steel containment vessel is via personnel hatches and a removable equipment hatch provides access for large component replacement. Internal containment atmosphere pressure and temperature is maintained by passive containment cooling (PCC) via an integral water tank situated in direct contact with the containment dome, providing passive cooling under accident conditions. Heat is removed from the hot steam and air inside containment via heat transfer through the containment dome structure to the water in the PCC tank located on the outside surface of the dome. Containment atmosphere response to a breach in the RCS may be characterized by two distinct phases. The first phase (blowdown) is an injection of hot steam corresponding to RCS depressurization. The high rate of steam injection during this period increases containment atmospheric temperature and pressure and quickly disperses a steam and air mixture throughout the containment volume. With cold walls and other structure surfaces, condensation is the most important heat transfer mechanism. Natural convection and conduction-limited heat transfer to the PCC tank distinguishes the second phase. There is sufficient water available to passively remove a minimum of seven days of core decay heat by evaporation.

6. Plant Safety and Operational Performances

Moderating and maximizing the time response of event loads relative to their limits is a focal point in improving the reactor inventory and cooling safety functions. The total inventory and its distribution throughout the system factor into this assessment. Further, reserve primary coolant from interfacing safety systems, most notably the RWST, can extend these time response periods both temporally and to a broader range of off-normal plant states. The arrangement of reactor core and steam generator thermal centers is crucial to the plant's capability to remove heat by natural circulation following a loss of forced circulation. By vertically separating these two components within an integral pressure vessel, the design of the mPower reactor encourages this natural convection heat removal rather than requiring engineered pump-driven systems. The mPower design includes analogous circuits to remove heat from the secondary and from the containment systems. In the former instance, the system is a non-safety, defense-in-depth system, providing the capacity for long-term decay heat removal. In addition, the reactor coolant inventory and purification system (RCI) serves as an active, non-safety decay heat removal system. System response derived from LOCA simulations has demonstrated that core temperatures are well below limits.

7. Instrumentation and Control systems

The instrumentation and control (I&C) system provides the capability to monitor, control and operate plant systems. It functions to (1) control the normal operation of the facility, (2) ensure critical systems operate within their designed and licensed limits, and (3) provide information and alarms in the control room for the operators. Important operating parameters are monitored and recorded, during both normal operations and emergency conditions to enable necessary operator actions. The I&C system is implemented using modern, scalable digital technology. Protection functions are implemented in a dedicated layer, which actuates engineered safety features if required to ensure safety of the facility. A second layer provides automatic control functionality and includes the capability for operators to control all systems within the plant. A third layer provides monitoring of all plant parameters, with advanced data processing capability to enable efficient operations. This layered architecture provides a high degree of automation while ensuring safety.

8. Plant Arrangement

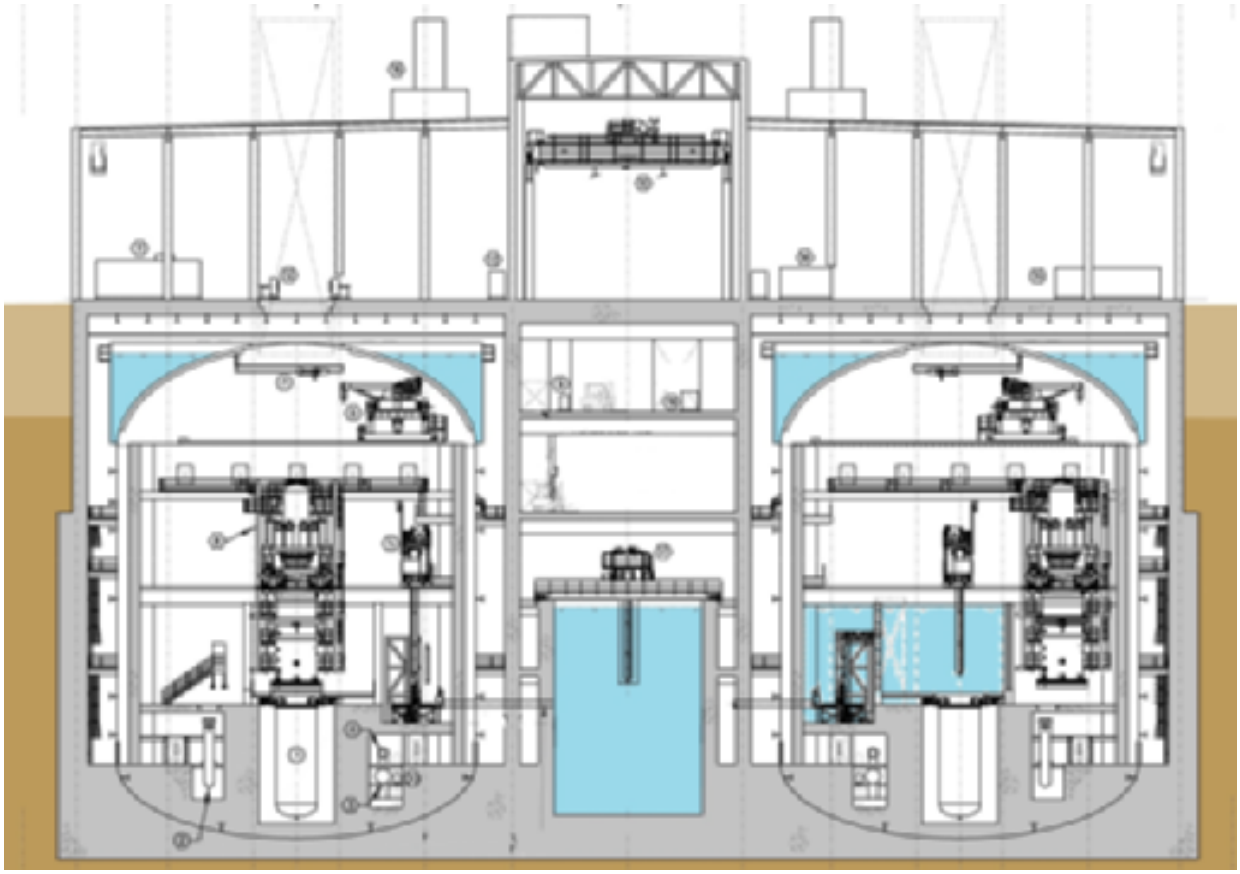


Figure 3: mPower plant underground containment vessel and spent fuel storage pool (Reproduced courtesy of Generation mPower LLC)

Reactor building

The reactor service building is a reinforced concrete, seismic Category I structure that surrounds the steel containment vessel and spent fuel storage pool which are located below grade level. The control room is located below grade in the reactor service building and contains the control system and operator interface for both reactors.

Balance of plant:

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. The BOP operation is not credited for design basis accidents. The steam and power conversion system is comprised of the turbine-generator, main steam system and condenser. Each reactor drives a separate turbine generator and no sharing of reactor safety systems. The design includes "Island Mode" capability to handle grid disconnection and is capable of 100 percent steam bypass capability to handle turbine trip, both preventing a need for a reactor scram.

Turbine Generator building

The turbine generators are housed in a separate building. The water cooled condenser provides for a nominal output power of 195 MW(e). The turbine-generator is designed for power manoeuvring and flexible grid interface. Turbine-generator support systems include a turbine bearing lubrication oil system, an electro-hydraulic control system, a turbine gland seal system, turning gear, over speed protective devices, a generator rectifier section and a voltage regulator.

Electric power systems

The main generator supplies power to plant auxiliaries during normal plant operation through an isolated phase bus duct and the unit auxiliary transformer. Offsite power to plant auxiliaries during startup, shutdown, and outage conditions is supplied via back-feed from the main transformer and the unit auxiliary transformer, with the generator circuit breaker open. If the unit auxiliary transformer is not available, offsite power is supplied via the station service transformer.

9. Design and Licensing Status

BWX Technologies, Inc. (formerly, The Babcock & Wilcox Company) and Bechtel Power Corporation are members in a formal alliance called Generation mPower LLC organized to design, license and deploy mPower modular plants. In 2013, the mPower program became the first recipient of funding under the US Department of Energy SMR Licensing Technical Support public-private cost-share program. Design engineering activities in support of a Design Certification Application continue at BWXT and Bechtel to further develop the technology with a focus on design certification. Design Certification and site specific licensing is expected to be completed in order to support an initial deployment in the mid-2020s.

Westinghouse SMR (Westinghouse Electric Company, LLC., United States of America)

1. Introduction

The Westinghouse small modular reactor (SMR) is an integral pressurized water reactor (PWR) design that builds upon 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 a net electrical output of greater than 225 MW(e) as a standalone 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.

Figure 1: Reactor System Configuration of Westinghouse SMR (Reproduced courtesy of Westinghouse Electric Company LLC)

2. Target Application

The target application is the clean and safe generation of electricity; however, the Westinghouse SMR also can be used to provide process heat, district heat, and off-grid applications, including the generation of power necessary to produce liquid transportation fuel from oil sands, oil shale, and coal-to-liquid applications.

3. Development Status

The Westinghouse SMR has been developed into a mature concept that relies heavily on proven and licensed technology previously developed by Westinghouse during deployment of other nuclear plants. The level of development is sufficient to perform detailed technical evaluations and give a timely response to market demand.

4. General Design Description

Design Philosophy

Design of the Westinghouse SMR utilizes passive safety systems and proven components – realized in the AP1000 plant reactor design and earlier Westinghouse designs – to achieve the highest level of safety, resiliency, and certainty in licensing, construction, and operations. The Westinghouse SMR is designed to be 100 percent modular and limits the size of primary components in order to enable unrestricted transportation, which reduces the need for costly infrastructure and increases the number of possible sites.

Nuclear Steam Supply System

The Westinghouse SMR utilizes a light water pressurized reactor coolant system (RCS) that is integrated into a single component, which eliminates the large-break loss-of-coolant accident (LOCA) from the postulated event.

Figure 2: Cut-Away View of the Westinghouse SMR Containment and Reactor Vessel (Reproduced courtesy of Westinghouse Electric Company LLC)

Reactor Core

The Westinghouse SMR reactor core is based on the licensed Westinghouse robust fuel assembly (RFA) design, and uses 89 standard 17×17 fuel assemblies with a 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 weight percent ^{235}U. Approximately 40 percent of the core is replaced every 2 years with the objective to achieve efficient and economical operating cycle of 700 effective full power days, which coincides with existing regulatory surveillance intervals.

Reactivity Control

Reactivity is controlled using the Westinghouse-developed system known as MSHIM™

control strategy or mechanical shim. MSHIM uses grey rods for short-term power control and boron dilution to adjust for fuel burnup over the longer term. Wireless instrumentation, comprised of hardened electronics and reactor control rod drive mechanisms (CRDMs) used in the Westinghouse SMR are based on proven AP1000 plant designs but modified to allow for placement within the harsh environment of the reactor pressure vessel (RPV). This proven design eliminates CRDM penetrations through the RPV head to prevent postulated rod ejection accidents, as well as the potential for nozzle cracking, which has negatively impacted currently operating plants. The upper internals of the RPV support 37 of these high-temperature-resistant, internal CRDMs for reactivity control during load-follow and similar operations.

Reactor Pressure Vessel and Internals

The RPV and reactor internals are designed to facilitate factory fabrication and shipment from the fabrication facility. Designed to meet the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, these components are derived from existing Westinghouse products but redesigned to function within the integral reactor assembly. The upper internals are an integral assembly containing all of the instrumentation and electrical penetrations to facilitate removal during refueling.

Reactor Coolant System

The Westinghouse SMR uses soluble boron in the reactor coolant for normal reactivity depletion, and control rods for load-follow and plant shutdown. The RCS is serviced by a number of auxiliary systems, including the chemical and volume control system, the normal residual heat removal system, the steam generator system, the primary sampling system, the liquid and gaseous radioactive waste systems, and the component cooling water system. Safety injection, passive boration, and heat removal are provided by the passive core cooling system and the ultimate heat sink (UHS) system.

Primary Cooling Mechanism

Both the integral reactor vessel and the passive core cooling system are located within the compact, high-pressure steel containment vessel located below grade. An equipment hatch is located in the top head of the containment vessel for maintenance activities. The containment vessel operates at a vacuum and is designed to be fully submerged in water to facilitate heat removal during accident events while providing an additional radionuclide filter.

Steam Generator / Pressurizer / Reactor Coolant Pumps

The Westinghouse SMR design incorporates eight seal-less canned motor pumps, which are mounted horizontally to the shell of the RPV just below the closure flange to provide forced reactor coolant flow through the core. A central primary riser directs the coolant flow as it exits the core to the steam generator. The reactor vessel downcomer acts as the channel for delivering the coolant flow from the reactor coolant pumps to the core inlet. The steam generator utilizes straight tubes with the primary reactor coolant passing through the inside of the tubes and the secondary coolant passing on the outside. An integral pressurizer is located above the steam generator within the RPV to control pressure in the primary system. The moisture separation functions typically performed in the steam generator occur in the SMR design in a separate steam drum located outside of containment, reducing the reactor and containment vessel heights by approximately 6 meters. The steam generator/pressurizer assembly can be removed for refueling operations through a bolted closure flange near the top of the integral reactor vessel.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	Westinghouse Electric Company LLC
Country of origin	United States
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)	65000
Coolant/moderator	Light water
Primary circulation	Forced circulation
System pressure	15.5
Core inlet/exit temperatures (^oC)	294/324
Main reactivity control mechanism	CRDM, boron
RPV height (m)	28
RPV diameter (m)	3.7
Configuration of reactor coolant system	Integral type
Power conversion process	Indirect Rankine Cycle
Fuel type/assembly array	UO_2 pellet/17×17 square
Fuel assembly active length (m)	2.4
Number of fuel assemblies	89
Fuel enrichment (%)	< 5
Fuel burnup (GWd/ton)	> 62
Fuel cycle (months)	24
Cogeneration capability	Possible
Approach to engineered safety systems	Passive
Number of safety trains	Three diverse decay heat removal
Refuelling outage (days)	17
Distinguishing features	Incorporates passive safety systems and proven components of the AP1000 plant and earlier Westinghouse designs
Modules per plant	1
Target construction duration (months)	18–24
Seismic design	Based on CEUS sites
Core damage frequency (per reactor-year)	< 5E-8
Design Status	Concept design completed

5. 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 alternating current (ac) power or other support systems to perform its safety functions. The 7-day minimum coping time following loss of offsite power is a fundamental advancement over the 3-day coping time applied in the operating plants. The integral reactor design eliminates large loop piping and potential large

break LOCA, and reduces the potential 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, thus 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 blackout.

Engineered Safety System Configuration and Approach

The Westinghouse SMR is designed with passive safety systems that utilize the natural forces of evaporation, condensation, and gravity. The design basis and licensing of passive systems were first implemented in the design of the AP1000 plant. Elements of these systems are described in the following sections.

Decay Heat Removal System

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. 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 (CMTs). Heat from the CMTs is then rejected to four heat exchangers located in two UHS system tanks. The UHS tanks are sized to provide a minimum of 7 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 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) that 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.

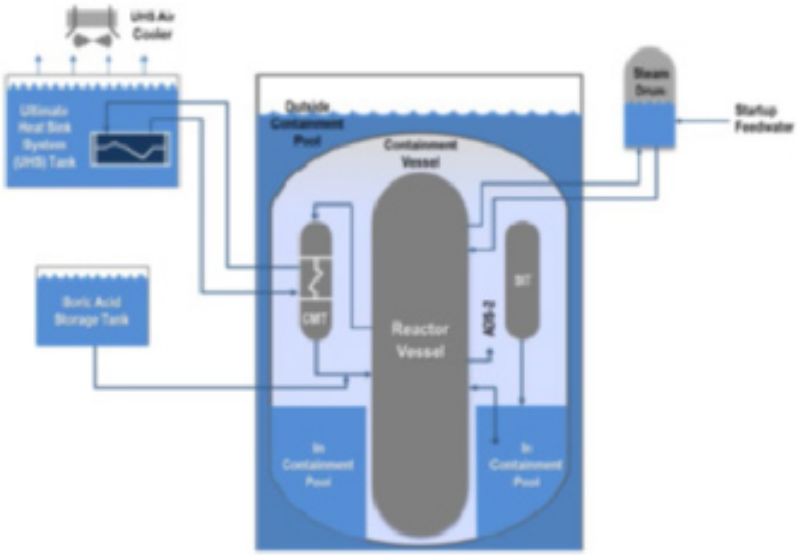


Figure 3: Three Diverse Decay Heat Removal Methods of the Westinghouse SMR (Reproduced courtesy of Westinghouse Electric Company LLC)

Containment System

The containment vessel is a carbon steel vessel that is normally submerged in a pool of water. Pressure in the containment vessel following postulated events is maintained by transferring heat through the shell to the water surrounding it. As this water boils, the inventory is made up from the two large UHS tanks that supply the plant with enough decay heat removal capacity for more than 7 days.

Figure 4: Below-Grade Location of Westinghouse SMR Reactor and Containment Vessels (Reproduced courtesy of Westinghouse Electric Company LLC)

6. Plant Safety and Operational Performances

The design of the Westinghouse SMR represents a significant advancement in plant safety with an estimated core damage frequency of 5E-8 per reactor year while maintaining an expected capacity factor of 95 percent.

7. Instrumentation and Control Systems

An **Ovation™**-based digital instrumentation and control (I&C) system controls the normal operations of the plant. The protection and safety monitoring system 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 controls non-safety-related components in the plant that are operated from the main control room or remote shutdown workstation. The diverse actuation system is a non-safety-related, diverse system that provides an alternate means of initiating reactor trip and actuating selected engineered safety features.

Ovation is a trademark or registered trademark of Emerson Electric Company. Other names may be trademarks of their respective owners.

8. Plant Arrangement

Reactor Building

The entire nuclear island is divided into five distinct and physically separate sectors all located below grade. The five independent sectors house a radiological control area that includes the reactor vessel, high-pressure containment vessel and spent fuel pool, and all safety-related systems, including safety-related dc power and I&C systems. No piping, wireways, or passageways traverse between these independent sectors. Access to each of the five independent sectors is instead regulated above grade. This design provides protection against external threats and natural phenomena hazards.

Figure 5: Plant Layout of Westinghouse SMR
(Reproduced courtesy of Westinghouse Electric Company LLC)

Reactor Control Room

The Westinghouse SMR main control room is also located completely below grade on the nuclear island; additionally, there are multiple security monitoring stations located in separate sectors.

Balance of Plant:

Turbine Generator Building

The electrical generator is designed for air cooling, which eliminates the potential for explosions that can occur with hydrogen-cooled options. The Westinghouse SMR condenser design includes the capability 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.

Electric Power Systems

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 composed of the independent Class1E 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 protection and safety monitoring system 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.

9. Design and Licensing Status

The Westinghouse SMR concept design has made substantial progress in support of U.S. and UK licensing. Westinghouse is considering a number of business models for the successful deployment of the Westinghouse SMR product globally. In addition, in February 2015, the U.S. Nuclear Regulatory Commission (NRC) approved Westinghouse's testing approach for the Westinghouse SMR design. The NRC approval is a significant step toward design certification and will reduce the time ultimately needed to license the Westinghouse SMR. In a letter dated February 27, 2015, the NRC told Westinghouse that it had granted a Safety Evaluation Report for the licensing topical report that the company submitted in April 2012 for agency review and approval. The topical report, developed by a panel of experts from inside and outside of Westinghouse, identified what would occur in the unlikely event of a small-break LOCA in the Westinghouse SMR. It also defined the test program that Westinghouse will conduct in the future to prove that its safety systems would safely shut down the reactor in response to a small-break LOCA. As a major technical innovation, the potential for intermediate and large-break LOCAs is eliminated in the Westinghouse SMR design because there are no large penetrations of the reactor vessel or large loop piping.

10. Plant Economics

Given that some elements of nuclear power plant design come with a relatively fixed cost regardless of power output, economy of scale challenges all SMR designs. To address the challenge, the Westinghouse design maximizes the power output for the material used in the design. Additionally, it is recognized that a significant contributor to the cost of nuclear is associated with time required to construct the plant and the cumulative cost of money during that time. To condense the construction schedule, the Westinghouse SMR is fully modular. Structures, systems, and components are manufactured in fabrication facilities and shipped to the site for assembly, allowing for extensive parallelization and schedule compression.

SMR-160 (Holtec International, United States of America)

1. 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 similar to existing commercial LWR fuel, 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.

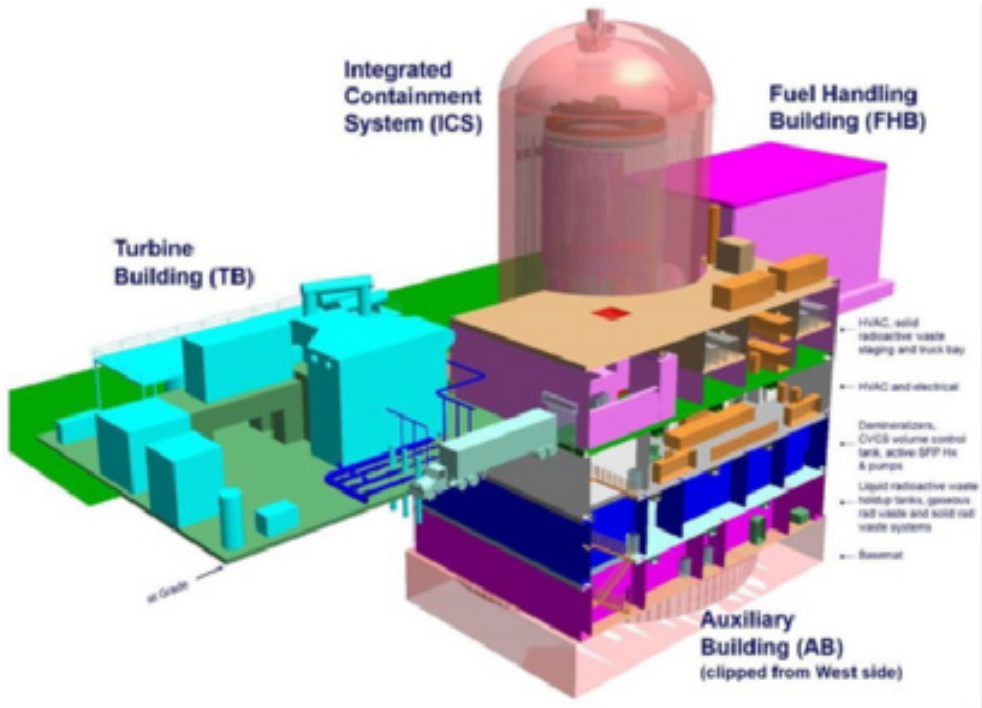


Figure 1: SMR-160 Nuclear Power Plant (Reproduced courtesy of Holtec International, Inc.)

2. Target Application

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 for 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. The plant design includes a unique fuel management solution that incorporates an integrated state-of-the-art underground dry fuel storage facility with the base plant offering.

3. Development Milestones

April 2015	Conceptual Design Complete
November 2017	Preliminary Design Complete
April 2018	PSAR/PSID Complete

4. General Design Description

Design Philosophy

The SMR-160 design is driven by the principal criterion that all safety significant systems must be powered by natural circulation. Core strengths and innovation of SMR-160 are its inherent safety, security, constructability and simplicity of operation.

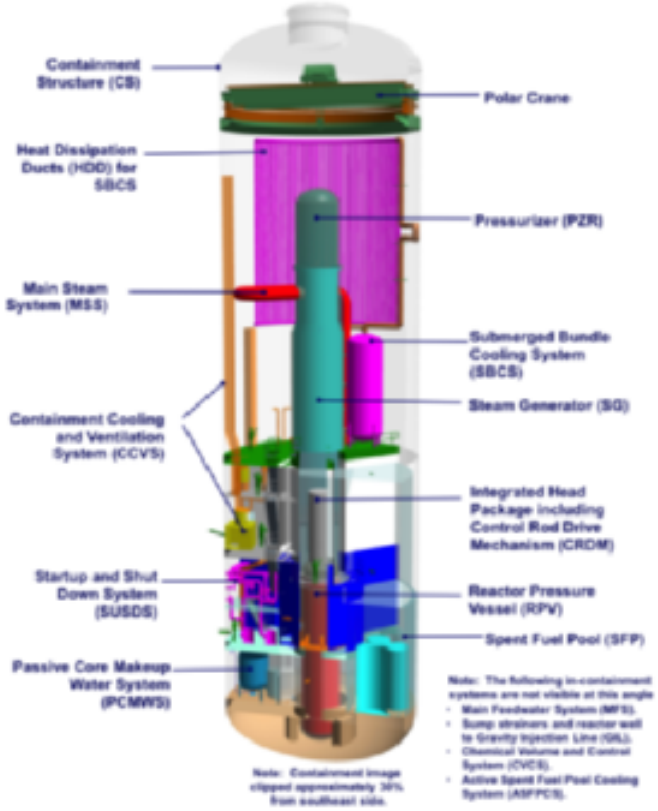


Figure 2: Detailed view inside SMR-160 containment
(Reproduced courtesy of Holtec International, Inc.)

Nuclear Steam Supply System

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 and trains of feedwater heaters, while not compromising the thermodynamic efficiency of the plant. The secondary side has only two feedwater heaters simplifying plant operations and maintenance

Reactor Core
The SMR-160 employs an efficient core design that uses traditional reload shuffle at the end of each cycle. The core is designed for up to a two year cycle with flexibility for shorter cycles depending upon utility requirements. Because of the relatively low number of fuel assemblies and overall outage schedule, the shuffling of nuclear fuel is not a critical path activity during an outage. The core contains an array of 96 square-lattice assemblies that are approximately 5.5 inches square and 14 ft. long in active height. Fuel bundles are surrounded by 16 reflector assemblies that contain pins of solid stainless steel or other reflecting material contained within stainless steel tubes. The reflector assemblies allow for an axial variation in the radial reflector by varying the composition of the pins along their length. The reflector assemblies can also be changed on a cycle to cycle basis if necessary depending upon fuel utilization requirements. Surrounding all of the fuel and reflector assemblies is a solid stainless steel reflector structure. This provides reflection of fast neutrons in order to improve the neutron economy, while reducing the fast neutron fluence through the reactor pressure vessel thereby, increasing vessel life.

Reactivity Control
Burnable absorbers integral to the fuel are used for long-term reactivity control and are designed to optimize 3D power distributions, cold shutdown margin, and hot excess reactivity. Short term changes in reactivity are controlled by movements of cruciform control blades, eliminating the need for using boric acid as a chemical shim. A core operated without chemical shim reduces operational costs and minimizes radioactive waste. In addition, boron-free operation increases reactor stability due to a strong negative moderator temperature coefficient of reactivity. The 21 cruciform control blades are very similar to BWR blades in that they contain an array of stainless steel tubes in each wing. The stainless steel tubes are filled with compacted B_4C powder that serves as the thermal neutron absorber. The control blades enter from the top of the RPV, eliminating penetrations in the bottom vessel head.

Reactor Pressure Vessel and Internals
The RPV is an ASME code III, Class 1, thick-walled cylindrical pressure vessel with an integrally welded bottom head and a removable top head. The upper extremity of the RPV shell is equipped with a tapered hub flange, which is bolted to a similar flange welded to the top head. The offset configuration of the SG and RPV also enables the use of traditional external control rod drive mechanism.

Reactor Coolant System
The SMR-160 RCS operates purely by gravity, as reactor coolant circulates entirely by virtue of the density difference in the primary water, which is present as long as fuel assemblies in the core produce heat. There are no reactor coolant pumps in the system. The RCS consists of three major components as shown in *Figure 4*: the (1) RPV and (2) SG with integral (3) Pressurizer.

Steam Generator
The SMR-160 has a vertical once through straight tube SG with the reactor coolant flowing inside Inconel 690 thermally treated tubes. The vertical straight tubes provide easy access for in-service inspection of the tubes. The SG uses sub-cooled feedwater to produce superheated steam on the shellside. The SG features a large inventory of secondary water on the shellside which provides substantial time margin to dryout (>4 hours).

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)	20500
Coolant/moderator	Light water
Primary circulation	Natural circulation
System pressure	15.5
Core inlet/exit temperatures (^oC)	196/316
Reactivity control mechanism	CRDMs
RPV height (m)	15
RPV diameter (m)	2.7
RPV or module weight (metric ton)	200
Configuration of reactor coolant system	Integral type
Power conversion process	Indirect Rankine Cycle
Fuel type/assembly array	UO_2 pellet/ square array
Fuel assembly active length (m)	3.7
Number of fuel assemblies	N/A
Fuel enrichment (%)	<4.95
Fuel burnup (GWd/ton)	45 GWd/t (initial design)
Fuel cycle (months)	24 months
Cogeneration capability	Possible
Approach to engineered safety systems	Passive
Number of safety trains	2
Refuelling outage (days)	10
Distinguishing features	• Unique approach to Defence-in-Depth with active non-safety and passive safety cooling systems • Integrated, indefinite, passive spent fuel cooling • Integrated spent fuel dry storage
Modules per plant	1

Pressurizer

The pressurizer is integral to the steam generator and engineered to maintain primary water and vapor interface to control pressure in the RCS. The pressurizer contains conventional heating and quenching elements (water/steam).

5. Safety Features

The SMR-160 safety basis incorporates defence-in-depth via multiple and diverse simple pathways for heat rejection from the core. All safety systems are within the containment structure, making them secure and safe from external threats. A large inventory of water

within a reservoir outside the containment structure provides long-term post-accident coping, and that structure facilitates air cooling for decay heat removal if the reservoir inventory boils off; for an unlimited coping period. SMR-160 engineered safety features include the passive core cooling system (PCCS), passive spent fuel pool cooling system (PSFPCS) and automatic depressurization system (ADS). A Holtec International innovative design; the PCCS and PSFCS are tied to the integrated containment system (ICS), which is the plant ultimate heat sink (UHS).

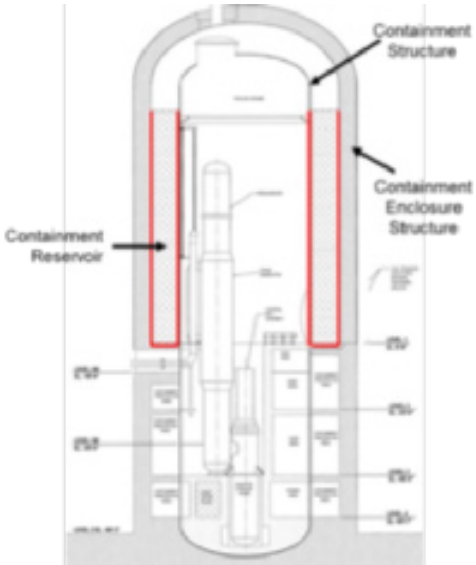


Figure 3: SMR-160 Integrated Containment System
(Reproduced courtesy of Holtec International, Inc.)

Integrated Containment System

The SMR-160 ICS consists of 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 (See *Figure 3*). An annular coolant reservoir (CR) between the CS and the CES is filled with water and serves as the UHS for SMR-160. In addition to preventing the release of radioactive fission products to the environment, the ICS acts as a large passive heat exchanger to cool the core and the spent fuel pool. The PCCS and PSFPCS use this system (via the CS metal surface) to reject heat to the environment. The reservoir provides sufficient inventory to remove the peak decay heat; as the water in this reservoir depletes (reservoir becomes dry > 60days), the ICS transitions to air cooling to remove subsequent decay heat. The operator has the option of replenishing the reservoir. The SMR-160 defence-in-depth architecture incorporates non-safety active systems for core and spent fuel pool cooling. The safety systems and non-safety systems are also configured to mitigate severe accidents.

Engineered Safety System Configuration and Approach

In the event of abnormal transients or postulated accidents, SMR-160 will employ non-safety active cooling systems as a first line of defence. Failure of those systems will actuate the passive safety systems and the plant will shut down and remain safely cooled for an unlimited period without the need for power, make-up water or operator actions, eliminating the need for safety related emergency diesel generators (EDG). Engineered Safety Systems are shown in *Figure 4*. The PCCS consists primarily of two sub-systems - the submerged bundle cooling system (SBCS) and the passive core makeup water system (PCMWS).

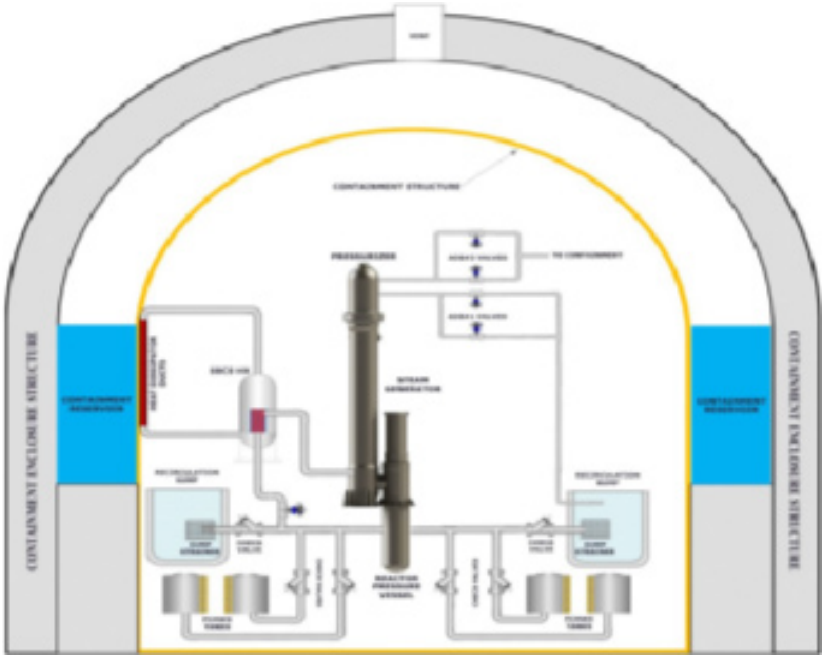


Figure 4: Schematic representation of SMR-160 engineered safety systems (Reproduced courtesy of Holtec International, Inc.)

Submerged Bundle Cooling System

For non-Loss-of-Coolant-Accident events (non-LOCA), the SBCS is actuated. This system is principally comprised of two heat exchangers – namely the submerged bundle heat exchanger (SBHX) and the heat dissipation Ducts (HDDs). In the event of an accident, primary coolant is routed to the SBHX, which is functionally equivalent to an auxiliary steam generator. The steam produced by this heat exchanger is routed to the CS via the HDDs, where it condenses by rejecting the latent heat to the containment reservoir through the steel CS. This system uses the elevation difference between the reactor and the SBHX to drive natural circulation flow of the primary coolant, and the elevation difference between the SBHX and HDDs to drive natural circulation flow of the secondary coolant. The SMR-160 also features an active non-safety residual heat removal system (RHR) for normal decay heat removal.

Passive Core Makeup Water System

In the event of a Loss-of-Coolant-Accident (LOCA), the PCMWS provides coolant inventory to keep the core covered at all times. The system provides medium pressure coolant injection using accumulator tanks (pressurized with nitrogen) for short term inventory make up and gravity injection for long term cooling. The coolant inventory lost via the break flashes into steam and liquid. The steam rises up and mixes with the containment atmosphere and condenses on the inside wall of the CS by rejecting its latent heat to the CR. The condensate and the entrained liquid are collected in the recirculation sump via gutters. Once the RCS reaches containment pressure, the coolant inventory in the sump reaches the core via gravity (gravity injection) due to the elevation difference between the sump and the core. For small break LOCA, the ADS actuates and the depressurization valves open (introduces an additional break) in order to depressurize the RCS, allowing gravity injection to keep the core covered. Gravity injection will commence once the RCS pressure equalizes with the containment pressure. In both sub-systems, the ICS is utilized as the heat exchanger (Ultimate Heat Sink) to passively reject decay heat to the environment.

Passive Spent Fuel Pool Cooling System (PSFPCS)

The SMR-160 spent fuel pool is located within containment. This feature not only contains

radioactive release from the spent fuel but also passively cools the spent fuel by tying to the ICS. By incorporating interim dry storage within the plant design, the spent fuel inventory in the plant never exceeds 250 assemblies. The plant design features both passive and active non-safety pool cooling systems.

6. Plant Safety and Operational Performance

The SMR-160 natural convection-driven reactor coolant loop is coupled with an optimized simple steam cycle, and is well adapted to load following

7. Instrumentation and Control systems

SMR-160 utilizes the Mitsubishi Electric Total Advanced Controller (MELTAC) platform as the backbone for the plant's I&C/HSI design. MELTAC – now proven with over 300 years of operating reactor experience - provides exceptional performance with attributes that facilitate a highly integrated plant-wide digital control system (DCS) configuration. MELTAC provides unique nuclear specific I/O and configuration flexibility to perform all nuclear safety and non-safety functions using the same digital platform. This includes Reactor Trip and Engineered Safety Feature actuation; integrated control functions such as feedwater, turbine and rod control; and I/O modules directly compatible with in-core and ex-core neutron monitoring detectors, and radiation monitoring detectors. The SMR-160 I&C/HSI minimizes plant capital cost, as well as long term operation and maintenance costs, by minimizing the number of platforms (two), safety divisions (two), controllers, sensors, processors, field cables, operators (two operators with the capability to operate multiple units from a single control room), and plant staff by eliminating most periodic safety system surveillance and including on-line condition monitoring. The HSI features a Large Display Panel (LDP) - a spatially dedicated continuously visible (SDCV) overview display of critical safety and power production functions. The HSI also includes a Safety Console (SC); Class 1E LCD touch screens, referred to as safety visual display units (S-VDU), provide SDCV display Type A (support for credited manual actions, if any) and Type B (critical safety functions) parameters. The operator console (OC) is comprised of safety grade and non-safety grade LCD touch screens; non-safety operational video display units (O-VDU) are multi-division displays with soft controls for all plant functions and components (non-safety and safety). S-VDUs are used to control all safety functions and components. I&C architecture includes a Plant Safety System (PSS), Plant Control System (PCS), Diverse Actuation System (DAS) and Information Technology System Interface (ITS). PSS elements include the Reactor Protection processor, Component Control processor, Safety Bus and S-VDU processor. A plant IT System interface for operators is accommodated in the control console design. Data from the PSS, DAS, and PCS is sent to the plant's IT System network via the PCS Gateway to support long term archive, O&M applications, and corporate wide information accessibility.

8. Plant Arrangement

Reactor building

The SMR-160 reactor building shares a common basement with the auxiliary building. Nearly half of the reactor building is underground and the reactor core is located at the very bottom of the ICS, protecting it from unauthorized access and missile impact. *Figure 5* provides a depiction of the plant layout.

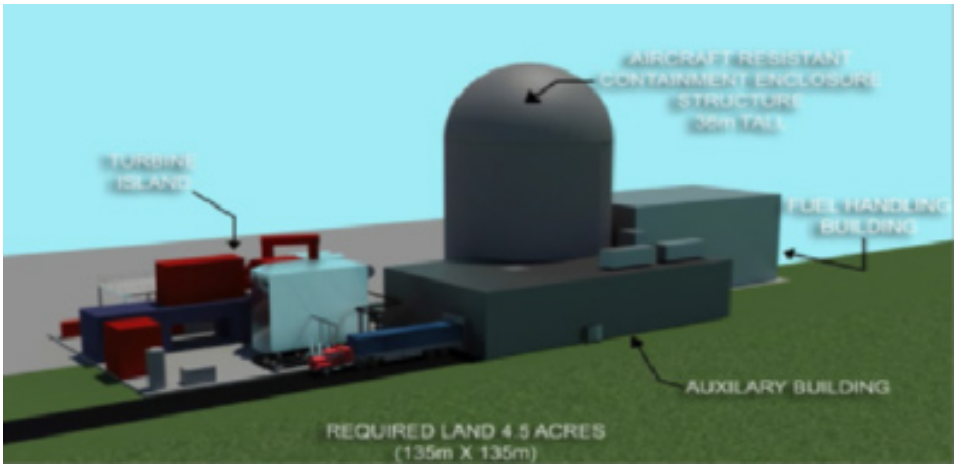


Figure 5: Plant Layout of SMR-160 (Reproduced courtesy of Holtec International, Inc.)

Control building
The SMART compact control room is designed for two man operation (staffing goal) under normal conditions of the plant. The main control room (MCR) is designed to provide air and cooling passively in any emergency situations.

Balance of plant:
Steam is produced at a relatively low pressure, 335 psi, but with considerable superheat, 170°F. The low pressure steam passes through an intermediate-low pressure turbine. The plant features only two feedwater heaters to supply sub-cooled feedwater to the steam generator.

Turbine Generator building
SMR-160 features an axial/side exhaust steam turbine with an option to use air cooled condensation. Superheated steam produced by plant is conducive for use in non-electric applications and also co-generation plants.

Electric power systems
The SMR-160 electric power system consists of the main generator, main transformer auxiliary transformers, non-safety diesel generators and safety related 1E batteries. There are two trains of Class 1E and non-Class 1E power supply. SMR-160 Class 1E power is required only for one-time alignment of valves for safety features actuation and post-accident monitoring. The Class 1E power supply derives power from physically separate battery banks.

9. Design and Licensing Status

Pre-application engagements are taking place with domestic and international regulators, and the Preliminary Safety Analysis Report is scheduled for completion by early 2018.

10. Plant Economics

Confidential plant economic data (capital cost, LCOE) have been developed and shared with, and are available to, interested commercial clients and stakeholders.

WATER COOLED SMALL MODULAR REACTORS (MARINE BASED)

ACPR50S (CGN, China)

1. Introduction

The ACPR50S is a small modular offshore floating reactor developed by the China General Nuclear Power Corporation (CGNPC) - aiming for high safety and adaptability, modularized design, and multi-purpose applications. It is intended as a potential optimal solution for combined supply of heat, electricity and fresh water for marine resource development activities, energy supply and emergency support on islands and along the coastal area.

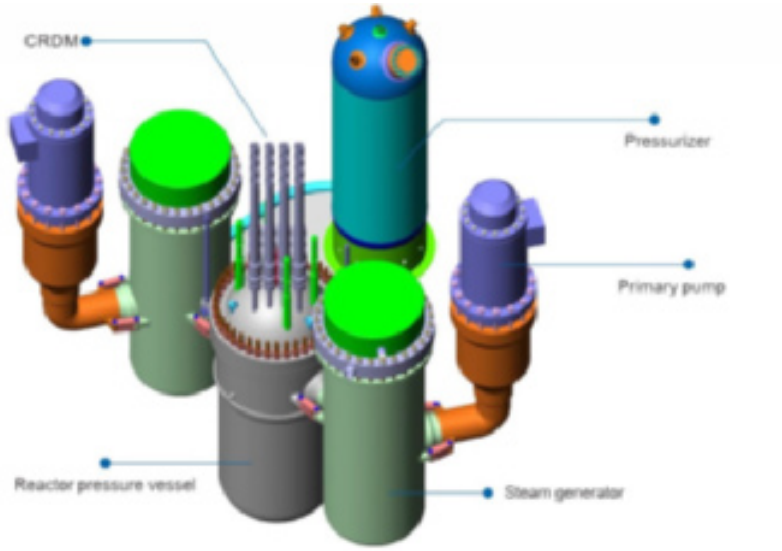


Figure 1: Reactor System Configuration of ACPR50S
(Reproduced courtesy of CGNPC)

2. Target Application

Being an offshore floating SMR, the ACPR50S is designed as a multipurpose power reactor for the following applications: combined energy supply for offshore oil drilling platform; offshore combined energy supply; coastland and island combined energy supply; energy supply for offshore mining, nuclear power ship; and distributed clean energy portfolio together with solar energy and wind power for islands. The ACPR50S has two (2) energy supply modes i.e. heat-electricity-fresh water-steam cascade supply mode and combined heat-electricity-fresh water-steam supply mode.

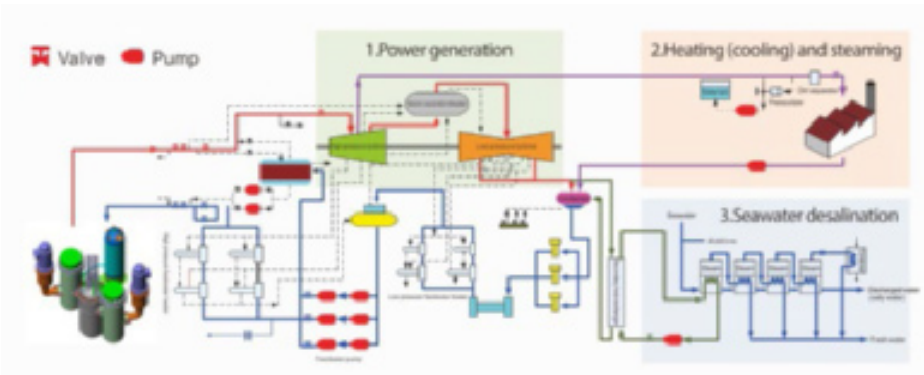


Figure 2: ACPR50S Energy Cascade Supply
(Reproduced courtesy of CGNPC)

Figure 3: ACPR50S for floating and fixed platform
(Reproduced courtesy of CGNPC)

Figure 4: ACPR50S for offshore combined energy supply station
(Reproduced courtesy of CGNPC)

3. Development Milestones

2009 – 2011	Conceptual study
2012	Starting Conceptual Design and formulating plan of theoretical tests
2013	Completion of Conceptual Design to determine the plan and major parameters
2014	Completion of Overall Design, preliminary demonstration and theoretical tests of the key systems; R&D on software; Starting study of Codes and Standards;
2015 – 2016	Preliminary Design Stages
2015	Completion of imported tests; completion of design software suitability evaluation; starting Preliminary Design; Completion of Codes and Standards study
2016	Completion of Preliminary Design and Preparation of Construction; Starting key equipment procurement for Project Construction Stage
2017 – 2020	Project Construction Stage
2017	Completion of PSAR review; Obtaining construction permit; Starting construction; and starting construction design
2018	Submission of FSAR and EIA; Completion of construction and installation; and obtaining fuel loading permit
2020 ~	Start up commissioning and connection to grid

4. General Design Description

Design Philosophy

The ACPR50S adopts design simplification with less cost and lower investment risks in order to be competitive with conventional offshore energy sources. Modular design is adopted through standardized streamline manufacturing aiming for shorter construction period as well as less cost. Higher load factor to be attained by a long refuelling cycle. ACPR 50S is capable of providing safe, clean, stable and efficient energy to satisfy various demands on heat, electricity, water and steam.

Nuclear Steam Supply System

The compact loop-type PWR nuclear steam supply system (NSSS) design of the ACPR50S consists of the reactor pressure vessel(RPV) that houses the core, two once-through steam generators (OTSG), two main coolant pumps(RCP) and a pressurizer(PZR) that are interconnected by short reactor coolant system legs. The compact layout of first loop reduces sharply the probability of LOCA. The primary cooling system is based on forced circulation during normal operation. The system has natural circulation capability under certain conditions, for example, under the condition of 10% thermal power and 14% flow rate, the natural circulation can be established.

Reactor Core

The low power density design with a slightly enriched UO_2 fuelled core ensures a thermal margin of greater than 15% which can accommodate any anticipated transient event. This feature ensures the core thermal reliability under normal and accident conditions. The fuel assemblies (FAs) of ACPR50S core with an axial length of 2.2 m core have a squared 17x17 configuration. The expected average fuel enrichment is less than 5% similar to standard PWR fuel. The reactor will be able to operate 30 months per fuel cycle.

Reactivity Control

The reactivity is controlled by means of control rods, solid burnable poison and soluble boron dispersed in the primary coolant. Burnable poison rods are introduced for flat radial and axial power profile, which results in an increased thermal margin of the core. The number and concentration of the burnable absorber rods in each fuel type are selected so that reactivity of each assembly can be as flat as possible. There are 16 control rods, with a magnetic force type control rod driving mechanism (CRDM).

Reactor Pressure Vessel and Internals

ACPR50S reactor pressure vessel envelopes and holds the core and the RPV internals, so that the fission reaction of the nuclear fuel is limited in one space. The RPV internals are used to support and fix the core component.

Reactor Coolant System

The ACPR50S primary cooling mechanism under normal operating condition and shutdown condition is done by forced circulation. The RCS has been designed to ensure adequate cooling of reactor core under all operational states, and during and following all postulated off normal conditions. The two RCPs are connected to the OTSG through short annular pipes, so are the two OTSGs to the RPV, therefore eliminated large bore piping and reduced opening of the main equipment. The integral design of RCS significantly reduces the flow area of postulated small break LOCA.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer	China General Nuclear Power Group (CGNPC)
Country of origin	China
Reactor type	Loop type PWR
Electrical capacity (MW(e))	60
Thermal capacity (MW(th))	200
Expected Capacity Factor (%)	95
Design Life (years)	40
Coolant/moderator	Light water
Primary circulation	Forced circulation
System pressure (MPa)	15.50
Core inlet/exit temperatures (^{o}C)	299.3/ 321.8
Main Reactivity Control Mechanism	Control rod driving mechanism (CRDM), solid burnable poison and boron solution
RPV Height (m)	7.2
RPV Diameter (m)	2.2
Module Weight (metric ton)	380
Configuration of Reactor Coolant System	Compact Loop (with short NSSS connecting pipes)
Power Conversion Process	Indirect Rankine cycle
Fuel Type/Assembly Array	UO_2/17x17 square pitch arrangement
Fuel Assembly Active Length (m)	2.2
Number of Fuel Assemblies	37
Fuel Enrichment (%)	<5%
Fuel Burnup (GWd/ton)	< 52
Fuel Cycle (months)	30
Cogeneration Capability	Possible
Approach to Engineered Safety Systems	Passive
Number of safety trains	2
Refueling Outage (days)	7 (to be defined later)
Distinguishing features	Floating power boat, once-through steam generator, passive safety system
Modules per Plant	Single
Target Construction Duration (months)	36
Predicted core damage frequency per reactor year	< 1E-6
Large release frequency per reactor year	< 1E-8
Design Status	Completion of conceptual/program Design, Preparation of project design

Steam Generator
The ACPR50S has two OTSGs with helically coiled tubes to produce superheated steam under normal operating conditions. The OTSGs are located on both sides of the reactor vessel. The small inventory of the secondary side (tube side) water in each OTSG prohibits a return to power following a steam line break accident. In the case of accidents, the OTSG can be used as the heat exchanger for passive residual heat removal system (PRHRS), which permits an independent operation of the PRHRS regardless of the hydraulic condition of the primary system.

Pressurizer
The pressurizer of ACPR50S is located outside of the reactor vessel connected to one of the hot leg connecting the RPV with a steam generator. The pressurizer is designed to control the system pressure at nearly constant level for the overall plant operation. The primary system pressure is maintained nearly constant due to the large pressurizer steam volume and the heater control. As the volume of the pressurizer is designed sufficiently large, condensing spray is not required for the load manoeuvring operation. The reactor over-pressure at the postulated design basis accidents related with a control failure can be reduced through the actuation of the pressurizer safety valve (PSV).

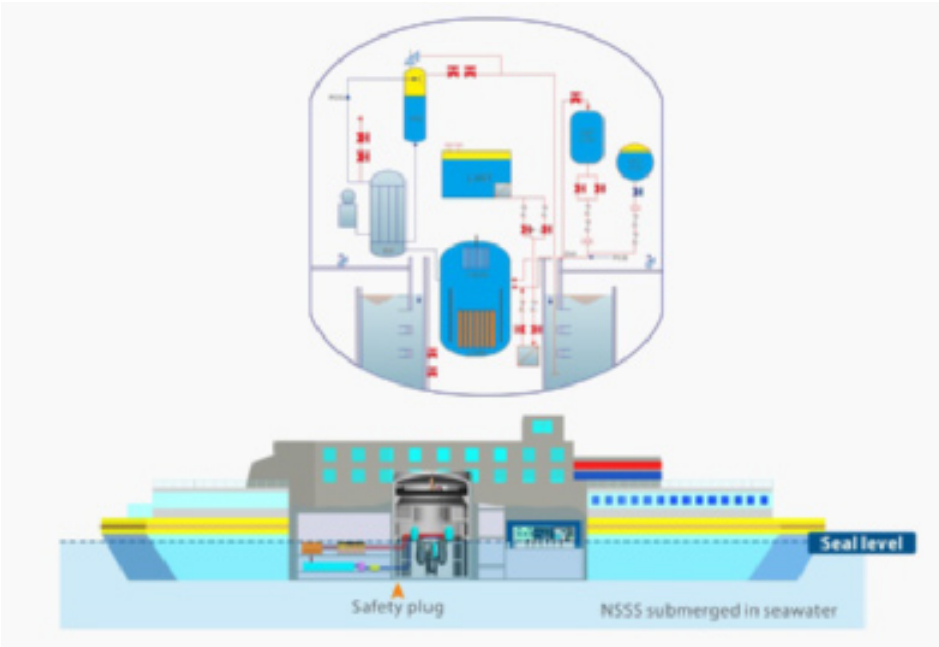


Figure 5: ACPR50S Reactor Building inside the Ship
(Reproduced courtesy of CGNPC)

5. Safety Features

The ACPR50S is designed with enhanced safety standards from the Generation III reactor designs to meet the requirements of national laws and regulation on the environmental release or radioactivity. A compact NSSS configuration with short nozzles will lead to reduced probability of LOCA. The NSSS in the ship is located under sea water level. The seawater is used as the ultimate heat sink and radiation shielding. Severe accident mitigation measures are incorporated to ensure low radioactive substance release probability to eliminate need for off-site emergency response. The ACPR50S is designed to cope with extreme external events such as typhoon, tsunami and ship collision.

Engineered Safety System Approach and Configuration

The ACPR50S is designed with passive safety systems which comprise passive safety injection system (SIS), automatic depressurization system (ADS), Passive Secondary Residual Heat Removal System (SHR), containment pressure suppression system (CPS), Passive Containment Heat Removal System (CHR), containment and containment isolation system (CIS), containment hydrogen control and filtration exhaust system (CHE).These passive safety systems are used to cope with design basis accidents (DBA) and severe accidents with core melts. The engineered safety system of the ACPR50S is shown in *Figure 6.*

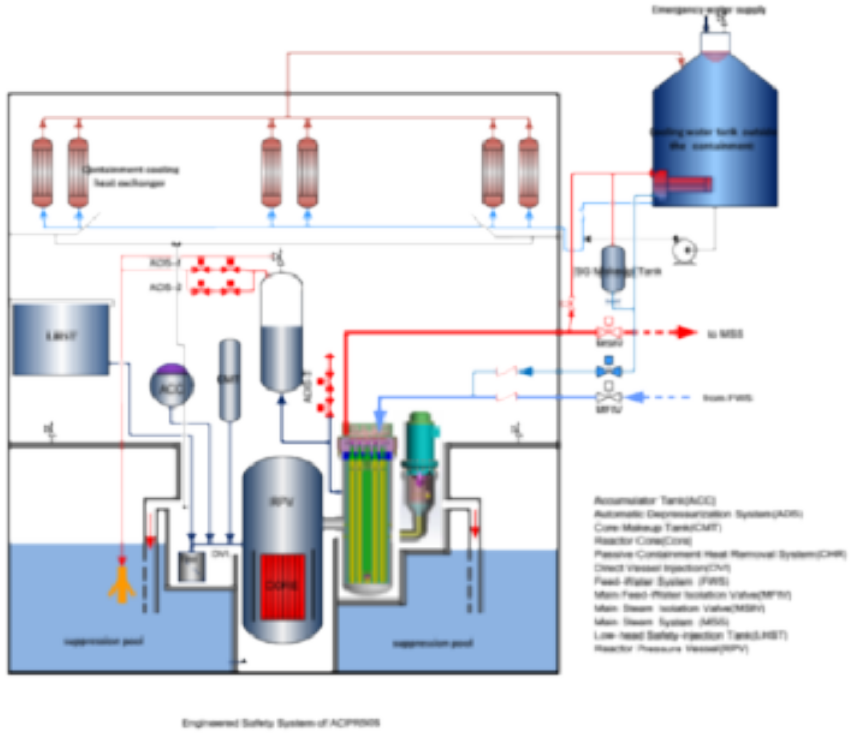


Figure 6: Engineered safety system of ACPR50S (Reproduced courtesy of CGNPC)

Decay Heat Removal System

The SHR is developed to remove the decay heat if the normal decay heat removal pathway is unavailable under accident condition. For non LOCA events, SHR removes the decay heat of the core through natural circulation between the OTSG and the SHR heat exchanger. And the decay heat is removed to the cooling water in the cooling water tank outside the containment eventually. In case of power supply is still available, if the normal decay heat removal pathway fails under non LOCA accident, the feed water system will be started instead of the SHR.

Emergency Core Cooling System

The SIS is a very important engineered safety system which is developed to cool the core and boride following DBA and extends the no operating time of plant operators to 7 days. Its main function is to control and mitigate the consequences of the accident to prevent the DBA from becoming serious accident t beyond design basis accident (BDBA). The core cooling is completely driven by the natural force following the DBA such as LOCA, which makes the system's composition and operation greatly simplified. Both high pressure and low pressure safety injection are driven by gravity, and the medium pressure safety injection is driven by compressed gas. In order to effectively connect the high pressure, middle pressure and low pressure safety injection, ADS is developed.

Containment System

The containment system is to contain radioactive material and to protect the environment against primary coolant leakage. The containment system of ACPR50S is actually a cabin called reactor cabin, a square containment. The containment has a volume of 870 cubic meters, a design internal pressure of 1.4 MPa and a design external pressure of 0.3MPa.The CHE is used to reduce the concentration of hydrogen in the containment to the safety limit under DBA and BDBA and to continuously monitor the hydrogen concentration at the top of the containment. The CPS is developed to cope with the DBA which can lead to the pressure rise inside containment vessel and suppress the containment pressure peak to ensure the integrity of the containment. The CPS is composed of the suppression pool system and suppression pool cleaning and cooling system. The containment heat removal system (CHR) is used to prevent the containment slow over temperature and overpressure. The CHR is a passive natural circulation system, and can provided 7 days of cooling for the containment in the case of no external water apply. The containment isolation system (CIS) is to provide isolation for the containment and to prevent and restrict the escape of radioactive fission products in the event of an accident.

6. Plant safety and Operational Performances

ACPR50S is suitable for all kinds of sea states such as swing, concussion, vibrancy, typhoon, seaquake and so on. The operational power range can be justified from 50% to 100% Pn (Power (nuclear)) and can be operated steadily for a long period for any power level of 50% to 100% Pn in order to satisfy the needs of users.

7. Instrumentation and Control systems

The I&C system design for ACPR50S will be based on defence in depth concept, compliance with the single failure criterion and diversity. The diversity in the design of I&C system is achieved through: The I&C system design for ACPR50S is mainly used in the steady-state and transient power during operation and provides automatic protection against unsafe reactors and abnormal operation, and provide the trigger signal to mitigate the consequences of accident conditions. Two reactors share one control room and one technical support center, a separate two remote shutdown stations, to ensure control and operation of the plant under normal and accident conditions.

8. Plant Arrangement

The single reactor module with the electrical, the steam generator, and nuclear auxiliary facilities is installed inside a non-propelled barge mounted ship for sharing facilities and reduced cost. Plant main building consists of the reactor containment cabin, the nuclear auxiliary cabin, the emergency diesel generator cabin and the turbine-generator cabin. For efficient radiation management, the plant main building is sub-divided into two zones, the duty zone and the clean zone. Systems linked with refuelling, overhauling, radwaste treatment are installed in the onshore basement.

Floating platform

The reactor building and fuel storage area are equipped with a full monitoring system with closed circuit monitoring system to oversee and prevent unauthorized access to the fuel. Reactor building is a pre-stressed concrete shell structure composed of a right cylinder with a hemispherical dome, with steel plate lining to act as a leak tight membrane. Reactor building is founded on a common base-mat together with the auxiliary building in which the main control room and fuel storage area are located. *Figure 7* shows the overview of the reactor cabin, auxiliary cabin, turbine-generator cabin, the emergency diesel generator cabin and the

main control room arrangement of ACPR50S.

Onshore basement

The onshore basement of ACPR50S houses the fuelling building, the radwaste treatment building, and other balance buildings of plant. Refuelling and overhauling will be performed in onshore basement.

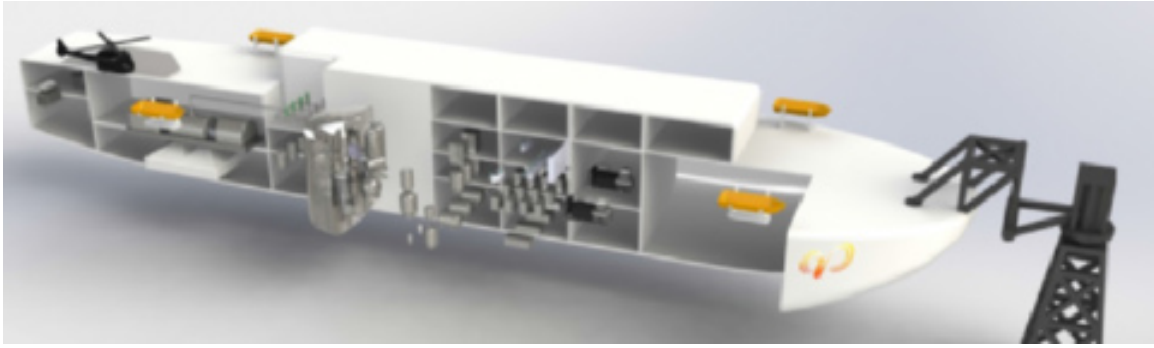


Figure 7: ACPR50S general plant arrangement in the float platform (Reproduced courtesy of CGNPC)

Control room

The compact control room is designed for one man operation under normal conditions of the plant. The main control room (MCR) is a key facility to cope with any emergency situations, so it is designed to ensure that plant personnel successfully perform the tasks according to the proper operating procedures. To achieve these goals, human factors engineering (HFE) process and principles are applied and verified using the full scope dynamic mock up.

9. Design and Licensing Status

The ACPR50S has completed the conceptual/program design work and is now preparing for project design. The National Nuclear Safety Administration（NNSA） is enacting the rule of safety review for small modular reactor. The ACPR50S plans to apply for the construction permit to the NNSA in 2017. An industrial demonstration plant of ACPR50S is being planned to be constructed in China, with a target of startup commissioning in 2020.

10. Plant Economics

The target plant construction cost of ACPR50S is about $7900/kWe and an electricity generating cost of ¢19/kWh.

Flexblue (DCNS, France)

1. Introduction

Flexblue is a transportable and immersed nuclear power system (TINPS), with an output capacity of 160 MW(e). The system is designed to deliver electricity using a Transportable & Immersed Nuclear Unit (TINU), later called "module" that is anchored with a positive floatability on the sea bed, between 40 m and a maximum depth of 100 m. Each module has the length of 146-meters and 14-meter diameter. Additional modules can be installed as demand increases. The nuclear units are positioned close to the shore and must be within the territorial waters. The reactor section is sealed in this configuration. A module encompasses a reactor (PWR with fuel enrichment classically less than 5%) and a turbine generating the electricity: it is manufactured in an assembly-line and assembled in a shipyard. System control and connection to the power grid are grouped together in a land-based facility called the "Shore Control Center" (SCC). They are connected to the immersed modules (or the farm of modules) via underwater buried cables. Fuel unloading and reloading and major maintenance operations are performed on a coastal base called the "Service Factory" (Supporting Site), which may be located at significant distances from the Operating Site (in the same country or not). *Figure 1* shows reactor system configuration.

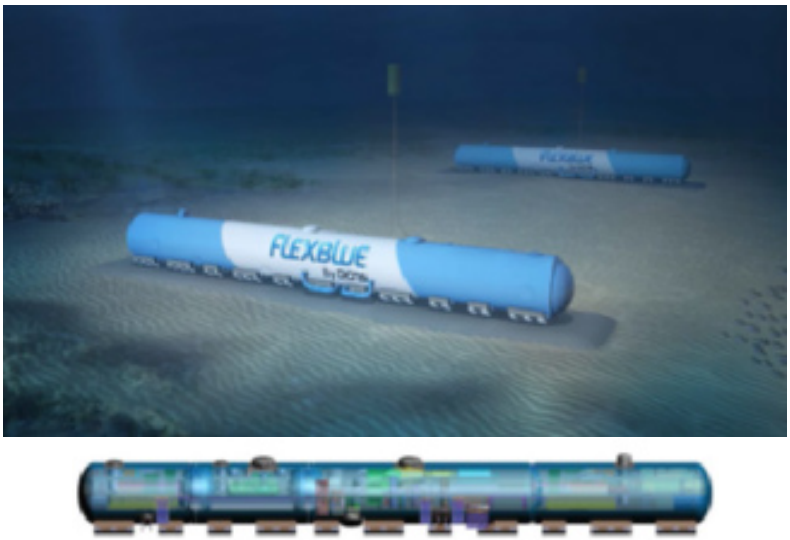


Figure 1: Reactor system configuration of Flexblue (Reproduced courtesy of DCNS)

2. Target Applications

Flexblue is designed to supply electricity to coastal grids. *Figure 2* shows the life cycle and arrangement of Flexblue that comprises two sites for Operating Site and Service Factory. The production cycle duration is equal to 3 years (in the Operating Site) and the theoretical availability over the 60 years of life-time is equal to 91%. After each cycle, the module is transported back to the Service Factory for major maintenance and refueling. Every ten years, a detailed inspection is also performed on this site, according to the international requirements. The Service Factory can be used and mutualized for many modules, whatever their owner and Operating Site origin.

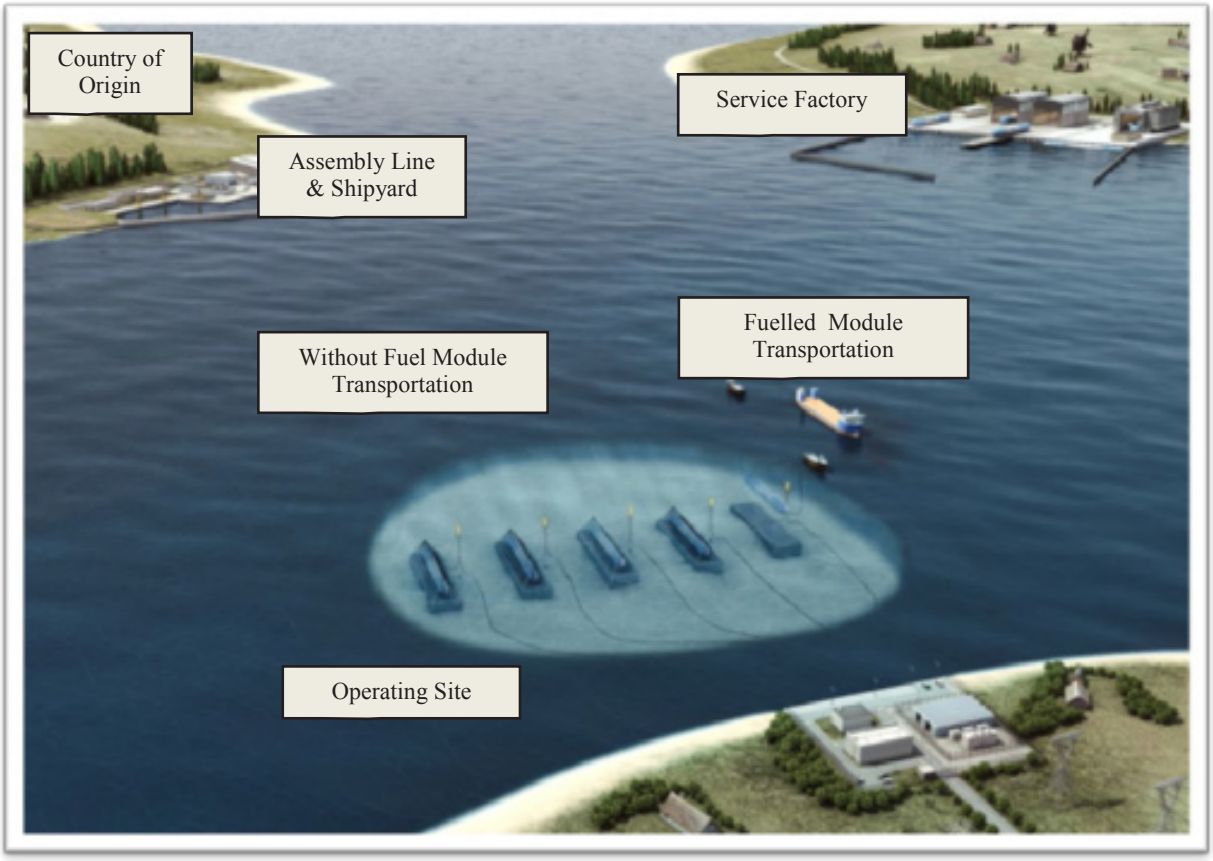


Figure 2: Flexblue Life Cycle and Arrangement (Reproduced courtesy of DCNS)

The following scenario is retained for the purpose of this document: these two installations are each located on the territory of two distinct countries. The life-cycle steps follow as shown in *Figure 2*; steps 4 – 5 – 6 – 7 – 8 are performed for the 60 years of the nuclear unit life-time:

1 – The module is assembled within the shipyard. Preliminary testing is performed without fuel.

2 – The module is transported to the Service Factory without any fuel.

3 – First fuelling and module trials are performed in the Service Factory. The first criticality is committed at that time.

4 – Module transportation with fresh fuel is carried out toward the Operating Site, throughout the international sea.

5 – Nautical operations are performed to position and to anchor the module on the sea-bed.

6 – The module delivers power to the grid via the Shore Control Centre. Minor maintenance is performed for the production operations in the Operating Site.

7 – Module transportation with spent fuel is carried out toward the Service Factory, throughout the international sea.

8 – It corresponds to the module's outage. Irradiated fuel is put within a devoted spent fuel pool in the Service Factory site and refuelling is carried out. Waste is removed from the module and major maintenance operations are performed.

By design (except when testing), the modules reactivity is authorized only when it is anchored and connected to the grid. Under all other circumstances, the reactor is shut down and criticality made impossible.

3. Development Milestones

2012 – 2016	The Project gathers international financial and operator stakeholders.

4. General Design Description

Design Philosophy

Flexblue design adopts proven technology that draw on DCNS's experience in nuclear propulsion and submarine power plants. The reactor design coupled with the concept of mooring it 100 m below the sea surface is envisaged to ease some of the safety issues of land-based nuclear power plants. The reactor is perceived to be resistant to natural disasters such as earthquakes, tsunamis and floods. The reactor will have the advantages of the ocean as a built-in coolant and ultimate heat sink.

Nuclear Steam Supply System

The module encompasses a 2-loop PWR with two horizontal steam generators. The Nuclear Steam Supply System (NSSS) operation diagram is presented below, in relation with the diverse configurations of the module.

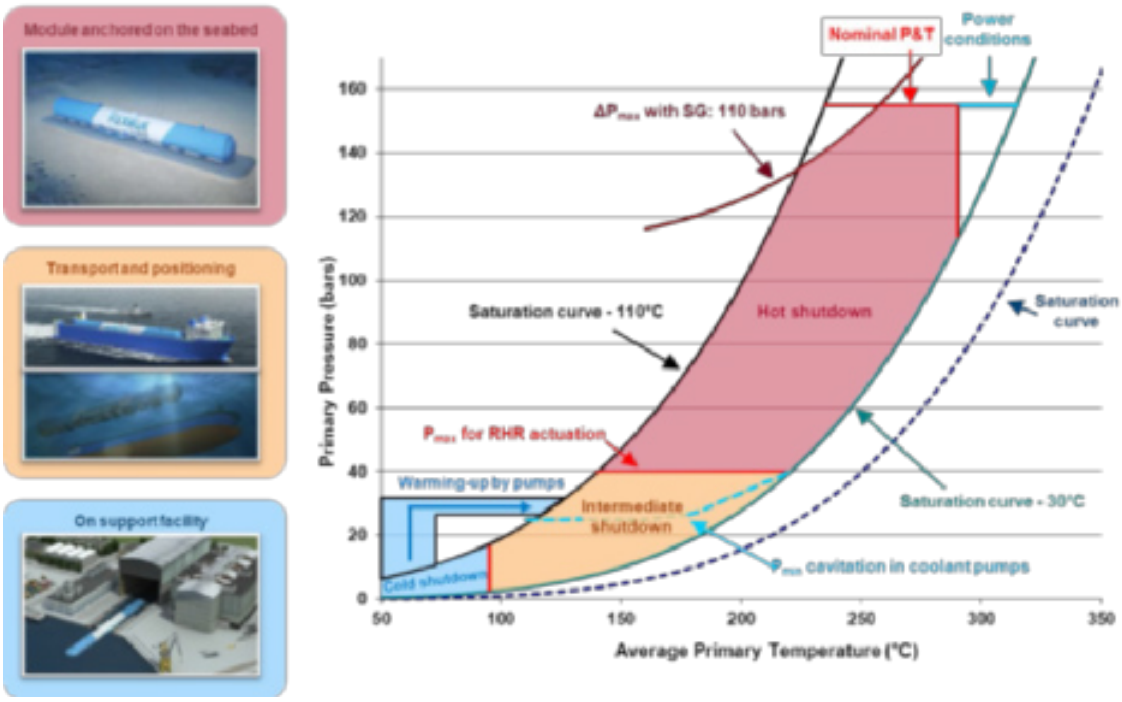


Figure 3: Reactor Operation Diagram of the module's NSSS
(Reproduced courtesy of DCNS)

Reactor Core

The reference core is made of 77 classical 17x17 fuel assemblies with an active length of 2.15 m. 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. Flexblue power production cycle lasts 38 months. At the end of a production cycle, the module is taken back to its support facility. The reactor is then refuelled and periodic maintenance is carried out. Major overhauls of the modules are scheduled every 10 years.

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 (%)	91
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
Core inlet/exit temperatures (^oC)	288/318
Main reactivity control mechanism	Control rods and solid burnable poison
RPV height (m)	7.65
RPV diameter (m)	3.84
Module weight (metric ton)	25000
Configuration of reactor coolant system	Loop type
Power conversion process	Indirect Rankine Cycle
Fuel type/assembly array	UO_2 and Zircaloy cladding/17x17 rods in square assembly
Fuel assembly active length (m)	2.15
Number of fuel assemblies	77
Fuel enrichment (%)	4.95
Fuel burnup (GWd/ton)	38
Fuel cycle (months)	38
Cogeneration capability	N/A
Approach to engineered safety systems	Passive
Number of safety trains	2 hydraulic trains and 2 x 3 I&C channels
Refuelling outage (days)	30
Distinguishing features	Transportable NPP, Submerged operation.
Modules per plant	Up to 6 per onshore main control room
Target construction duration (months)	36
Seismic design	Not comparable – subsea conditions
Predicted core damage frequency (per reactor year)	< 1 E-7
Design status	Building Partnership

Reactivity Control

A soluble boron free core with enrichment less than 4.95% is used. The absence of boron reduces the amount of produced waste and improves the safety in comparison with other reactors (no dilution accident, exclusion of control rod ejection, improved margins on boiling crisis, low power density, xenon oscillation stability). However, the boron absence leads to difficulties to reach a high burn-up; the core is surrounded by a heavy reflector. Basically, a

compromise has been adopted between availability, fuel cost and the core size. Furthermore, to avoid any dependency from the fuel supplier, standard square fuel assemblies are used. Those ones can be provided by any suppliers and no front end cycle dependency is introduced in this way.

Reactor Coolant System

The reactor adopts a proven loop-type PWR nuclear steam supply system. The primary cooling mechanism under normal operating condition and shutdown condition is by forced circulation of coolant. The reactor coolant system (RCS) is designed to ensure adequate cooling of reactor core under all operational states, and during and following all postulated off normal conditions. Sea water provides a permanent cooling of the reactor core and acts as the ultimate heat sink.

5. Safety Features

The aim of the design is to make impossible significant radioactive releasing on the Operating Site when the nuclear unit is anchored on the sub-marine site. The nuclear unit submersion during the electricity-production phase represents a clear advantage. The safety level of the module is greater compared to most of land-based facilities, small or large:

- Passive systems associated with the infinite heat sink which the sea provides;
- Inherent protection against numerous external hazards above the sea (natural or anthropic) due to the immersion and the water column (aircraft impacts, typhoons and other meteorological effects, etc.);
- Reduced intensity of tsunami and earthquake under the sea, in comparison with Land Based Reactors.

Furthermore, in case of implausible severe accident leading to core meltdown, the defense in depth is based on the In-Vessel Corium retention (I-VCR). The reactor residual thermal power (the heat decay) is smaller than any large reactors. It allows considering the external vessel cooling is sufficient to avoid any vessel rupture.

Engineered Safety System Approach and Configuration

The reactor integrates diversity in safety systems and segregates the operating stations to achieve a high level of safety. Water offers a natural protection against many of the possible external events and guarantees a permanently available heat sink, and an additional barrier to fission products in the case of an accident. The submerged Flexblue reactor has a minimal environmental footprint. 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. 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 positive buoyancy of the submerged unit allows 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. If a safety issue developed with the reactor, it could be brought to the surface and taken to a shipyard for repair. It could be refuelled in the same way, and at end of life it would be repatriated to the shipyards for decommissioning. The emergency power sources (batteries) are sized to ensure the autonomy of the modules up to 15 days without any external power. The safety of the plant is improved considerably with

four (4) operating areas in accident conditions: the main control room, the main remote shutdown station, the local control room and the local remote shutdown station.

Decay Heat Removal System

Decay heat removal is performed by four cooling loops, each one able to remove 50% of decay heat:

- Two primary chains are connected to the primary circuit: each one includes a heat exchanger immersed in a large safety water tank which is cooled by the ocean through the metallic hull.
- Two secondary chains are connected to the secondary circuit: each one includes an emergency condenser directly immersed in seawater.

With the infinite heat sink – seawater – and to the elevation difference of the heat sink with respect to the heat sources, the four chains operate passively by natural circulation. The long-term safe state of the reactor is a shutdown state where continuous cooling of the reactor core is achieved by natural circulation.

Emergency Core Cooling System

Protection against loss-of-coolant accidents is ensured by a passive safety injection system including direct vessel injection lines, core makeup tanks, accumulators and large safety tanks. All these injection sources inject passively into the vessel In addition, a two-train automatic depressurization system is connected to the pressurizer. Once these systems have actuated, a passive recirculation path is in place. Decay heat is removed through the metallic hull (the containment). In case of an accident, active systems designed for normal/shutdown core cooling or for controlling coolant inventory are used if alternating-current power is available. If not, the passive safety systems described earlier 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 valves and communication (data transfer from the module to the remote Control Room). The fourteen (14) days of autonomous monitoring ability can be extended by reloading the batteries. An emergency communications system works via radio links in case of loss of cables. Also, an emergency ultimate system can be run via an acoustic link, when losing both the submarine cables and the microwave link. This enables to communicate with a ship/submarine and transmits the module state and enables communications.

Containment System

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

6. Plant safety and Operational Performances

The reactor is based on the PWR technology and significantly benefits from a large operating experience in commercial power plants as well as in the naval environment. The concept adopts a tele-operation which is a great multiple stakes issue that contributes to the optimization of the life cycle cost, accident response and management, and ensure a good availability factor. The module is accessible via a submarine vehicle that connects to access

hatches, so that maintenance, inspection and operation can be performed on-board while on the seafloor. The refueling takes place approximately every 3 years. The module is removed and transported back to a coastal facility, which hosts the spent fuel pool. Major overhaul occurs every 10 years, i.e., every three fuel-cycles.

7. Instrumentation and Control Systems

The I&C and communications systems are based on different technologies, following the safety levels in conformity with the IEC standards. Several possible solutions to allocate the classes of functions to the controllers are considered with different technological and functional diversification schemes. I&C system is robust and incorporates experience from naval industry, especially from nuclear submarines. The I&C system adopts redundancies of trains, different levels of defence especially on communication systems, technological diversification and the principle of reactor unit I&C autonomy. The I&C system includes a diversified operating station (DOS) enabling commands of safety functions in case of technological failure on the nominal station. This diversified system works diversified solutions from the instrumentation to the man-machine interface (MMI). The main principles of I&C systems are based on module autonomy towards the shore control facility. The module autonomy is ensured by the fact that: all the systems supporting the safety functions are in the module; the reactor can be operated from two different places in the module during accident conditions; the transition from the initial state to a safe shutdown state when losing submarine cables is automatic; Local emergency power source. The concept also includes the possibility of multi-units operations from a single control room in order to run a cluster of modules. This aims to reduce the operating crew size considerably. Flexblue employs redundant main and auxiliary submarine cables that transport electricity as well as information between the modules and the onshore control centre.

8. Plant Arrangement

The module is composed of a turbine & alternator section, a reactor section, an aft section and a fore section. These 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. Several units can operate on the same site and hence share the same support systems.

9. Design and Licensing status

Presentations of the concept have been made to the French safety authority. Technical discussions have been initiated with the French technical safety authority. Preliminary conclusions have been drawn: these ones lead going ahead the project.

KLT-40S (OKBM Afrikantov, Russian Federation)

1. Introduction

The KLT-40S is a PWR developed for a floating nuclear power plant (FNPP) to provide capacity of 35 MW(e) per module. The design is based on third generation 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 nuclear power plant (NPP). The FNPP with a KLT-40S reactor can be manufactured in shipyards and delivered to the sites fully assembled, tested and ready for operation. There is no need to develop transportation links, power transmission lines or the preparatory infrastructure required for land based NPPs, and there is a high degree of freedom in selecting the location for a FNPP as it can be moored in any coastal region.

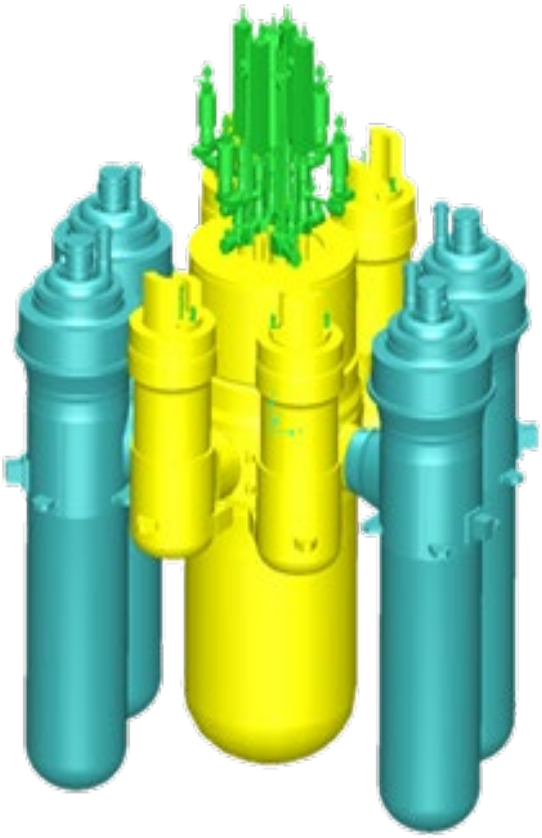


Figure 1: Reactor System Configuration of KLT-40S
(Reproduced courtesy of OKBM Afrikantov)

2. Target Application

The FNPP with KLT-40S is intended to provide cogeneration capabilities for power and heat supply to isolated consumers in remote areas without centralized power supply. Besides, this FNPP can be used for seawater desalination as well as for autonomous power supply for sea oil-production platforms.

Figure 2: Possible floating power unit configuration
(Reproduced courtesy of OKBM Afrikantov)

3. Development Milestones

1998	The first project to build a floating nuclear power plant was established
2002	The environmental impact assessment for KLT-40S reactor was approved by the Russian Federation Ministry of Natural Resources
2006	After several delays the project was revived by Minatom (Russian Federation Ministry of Nuclear Energy)
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. JSC "Baltiysky Zavod" (Saint Petersburg) undertook charge of construction, installation, testing and commissioning the first floating plant named Akademik Lomonosov (2 units with the output of 35 MW(e))
2017	Completion of construction and testing of the floating power unit at the Baltic shipyard
2019-2020	Commercial startup

4. General Design Description

Design Philosophy
KLT-40S is the reactor for FNPP Akademik Lomonosov, intended for reliable power and heat supply to isolated consumers in remote areas without centralized power supply and where expensive imported fuel is used.

Nuclear Steam Supply System
The steam lines while exiting from the SGs are routed through containment to a set of steam inlet 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.

Reactor Core
Fuel utilization efficiency is achieved by using dispersion fuel elements. One of the advantages foreseen by the FNPP under construction is long term independent operation in remote regions with decentralized power supply. The design requires refueling of reactor after every 3–4 years of operation. Refueling is performed 13 days after reactor shutdown when the levels of residual heat releases from spent FAs have reached the required level. The spent

fuel will be delivered to the importing country. The spent nuclear fuel will then be stored on board the FNPP. No special maintenance or refueling ships are necessary. Single fuel loading is done in order to provide maximum operation period between refueling. The fuel is loaded in the core all at once with all fuel assemblies being replaced at the same time. Besides, low-enriched U_{235} with enrichment below 20% is used to meet non-proliferation requirements.

Reactivity Control

The control rod drive mechanism (CRDM) is electrically driven and releases control and emergency control rods into the core in case of station black out (SBO). The speed of safety rods driven by electric motor, in the case of emergency is 2 mm/s. The average speed of safety rods being driven by gravity is 30 – 130 mm/s.

Reactor Pressure Vessel and Internals

The KLT-40S reactor has a four-loop forced and natural circulation coolant loop; the latter is used only in the emergency heat removal mode. This reactor is utilized at all operating nuclear icebreakers.

Reactor Coolant System

The reactor has a modular design with the core, steam generators (SGs) and main circulation pumps connected with short nozzles. The reactor has a four loop system with forced and natural circulation, a pressurized primary circuit with canned motor pumps and leak tight bellow type valves, a once-through coiled SG and passive safety systems. KLT-40S thermal-hydraulic connections comprising external pressurizer, accumulators, and separation heat ex-changer are in proximity of the reactor systems. The pressurizer is not an integral part of the reactor systems and in this design it is formed by one or more separate tanks, designed to accommodate changes in coolant volume, especially severe during reactor start-up. The core is cooled by coolant flowing from core bottom to top, in accordance with typical PWR core flow patterns. However, flow patterns between the core shroud and the RPV inner walls differ significantly from conventional external loop PWR configurations. Once hot coolant exits the top of the core and enters any of the multiple SGs, it uses coaxial hydraulic paths wherein the cold and hot legs are essentially surrounding one another. As hot coolant enters the SG, it begins to transfer thermal energy with the fluid circulating in the secondary loop (secondary side of the SGs).

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology Developer	Afrikantov OKBM
Country of origin	Russian Federation
Reactor type	PWR
Electric power (MW(e))	35
Thermal power (MW(th))	150
Expected Capacity Factor (%)	60 to 70
Design Life (years)	40
Coolant/moderator	Light water
Primary circulation	Forced circulation
System pressure (MPa)	12.7
Core inlet/outlet temperatures (^{o}C)	280/316
Steam capacity (t/h)	240

Overheated steam pressure at SG outlet (MPa)	3.82
Overheated steam temperature at SG outlet (^oC)	290
Main Reactivity Control Mechanism	Control Rod Drive Mechanism
RPV Height (m)	4.8
RPV Diameter (m)	2.0
SGU weight (without primary and secondary coolant) (t)	211
Configuration of the Reactor Coolant System	Compact Loop Type
Power Conversion Process	Indirect Rankine Cycle
Fuel Type/Assembly Array	UO_2 pellet in silumin matrix
Fuel Assembly Active Length (m)	1.2
Number of Fuel Assemblies	121
Fuel Enrichment (%)	18.6
Fuel Burnup (GWd/ton)	45.4
Fuel Cycle (months)	30-36
Cogeneration Capability	Possible
Approach to Engineered Safety Systems	Active (Partially passive)
Number of safety trains	2
Refueling Outage (days)	30 – 36
Distinguishing features	Floating power unit for cogeneration of heat and electricity; Onsite refuelling not required; Spent fuel take back to the supplier
Modules per Plant	2
Generator terminal power (MW(e))	2 x 38.5 (gross)
Maximum thermal power (Gcal/h)	2 x 73
Coastal area (ha)	1.5
Water area (ha)	6.0
Number of operating personnel (persons)	< 60
Target Construction Duration (months):	48 – 60 (including manufacturing of components)
Seismic design	9 point on the MSK scale
Predicted core damage frequency (event/reactor year)	0.5E-7
Length of the ship (m)	140
Width of the ship (m)	30
Hull height (m)	10
Full load weight (ton)	21000
Cogeneration plant service period (year)	40
Design Status	Under construction, planned commercial start 2019-2020

5. Safety Features

The KLT-40S is designed with proven safety aspects such as a compact structure of the SG unit with short nozzles connecting the main equipment, primary circuit pipelines with smaller diameter, and with proven reactor emergency shutdown actuators based on different operation principles, emergency heat removal systems connected to the primary and secondary circuits. Additional barriers are provided to prevent the release of radioactivity from the FNPP 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 the reactor emergency shutdown, emergency heat removal from the primary circuit, emergency core cooling and radioactive products confinement. The KLT-40S safety concept encompasses 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 FNPPs are distinctive from those applied to land-based installations in security of the water areas surrounding the FNPP, 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.

Engineered Safety System Approach and Configuration

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

Decay Heat Removal System

The decay heat removal system is intended to remove core residual heat upon actuation of reactor emergency protection in case of abnormal operation including accidents, as well as to remove residual heat at normal RP decommissioning. The decay heat removal system includes the following cooling channels:

- Two secondary passive cooling channels via steam generators;
- One active secondary cooling channel via steam generators;
- One active cooling channel via the primary/third heat exchanger.

Reactor core heat is removed by two passive system channels at de-energization for more than a day.

Emergency Core Cooling System

The Emergency Core Cooling System (ECCS) is intended to:

- Supply water to the reactor for core cooling in accidents associated with primary coolant loss;
- Makeup of primary coolant during process operations;
- Supply of liquid coolant to the reactor at failure of the electromechanical reactor shutdown system;
- Adjustment of water chemistry;
- Hydraulic testing of the primary circuit and associated systems, secondary and third loop sections disconnected at inter-circuit leaks and designed for primary pressure.

The ECCS includes the following subsystems:

- High-pressure ECCS subsystem with makeup (low-capacity high-pressure) pumps;
- High-pressure ECCS subsystem with hydraulic accumulators;
- Low-pressure ECCS subsystem with recirculation pumps.

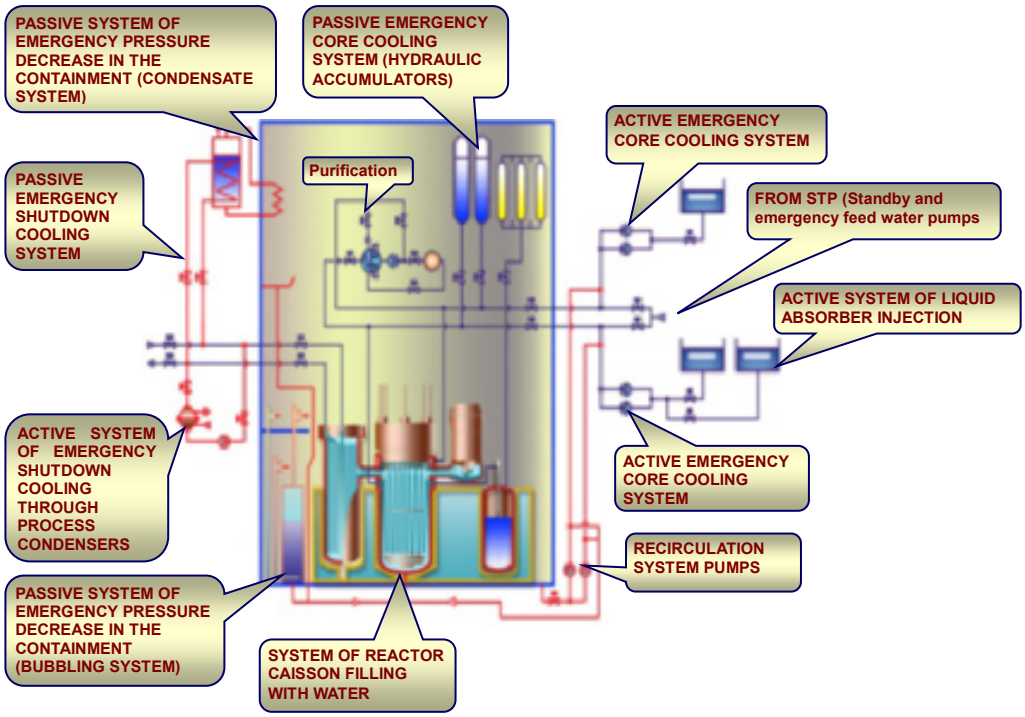


Figure 3: KLT-40S Emergency Core Cooling System flow diagram (Reproduced courtesy of OKBM Afrikantov)

Containment system

The containment for the KLT-40S is configured for FNPP applications and is made of steel shell designed to sustain mild pressurization, while the reactor systems are positioned inside a reinforced "reactor room" whose bottom forms a steel-lined tank. This tank can be flooded with cooling water for decay heat removal as well as for shielding purposes. The top portions of the reactor room can be pressurized as the reactor room is plugged by a steel and concrete plug. Once removed, the plug provides access to the reactor systems and to the core for refueling or maintenance operations.

6. Plant Safety and Operational Performance

The KLT-40S NPP ensures electricity and heat generation within the power range of 10% to 100% for a continuous operation of 26000 h. The NPP is designed for manoeuvring speed of up to 0.1%/s. As a countermeasure against the external impact, the NPP is fitted with both ground safety and floating physical protection means. Structures are designed to be placed in the Arctic zone at the depth of 2 m at freezing temperatures. The Floating Power Unit (FPU) and NPP buildings are designed to withstand the crash of an aircraft of 10 t. Based on the analysis of emergency situations, the radiation-ecological impact upon the personnel and environment at normal operation, abnormal operation, including design-basis accidents does not result in exceedance of established dose rates for the personnel and population, the content of radioactive substances in the environment, and it is limited at beyond-design-basis accidents.

7. Electric Power System

The electric power system in the FPU is comprised of the following: main electric system; and emergency electric system.

The main electric system of the FPU is intended to generate electricity and transmit it to the power system of the region, as well as to transmit electricity to internal consumers. The system includes two main three-phase AC generators of 35 MW each and eight back-up

diesel generators of 992 kW each. The emergency electric system supplies electricity to safety system loads in all operation modes, including loss of operating and back-up electric power sources. The FPU has independent emergency electric systems for each reactor plant. Each emergency electric system has two channels with an emergency diesel generator of 200 kW.

8. Plant Arrangement

The coastline line of the Floating Cogeneration Nuclear power Plant (FCNPP) has the following auxiliary facilities:

- Complex engineering building with equipment to distribute and transmit electricity to loads and to prepare and transfer heating water to loads;
- Auxiliary buildings, including: two hot water storage tanks; partially in ground tank with slime water; wet storage bunker; two cooling towers; access control point; site enclosure; lighting towers.

The coastal line of the FCNPP does not provide for handling nuclear materials and radiation hazardous media.

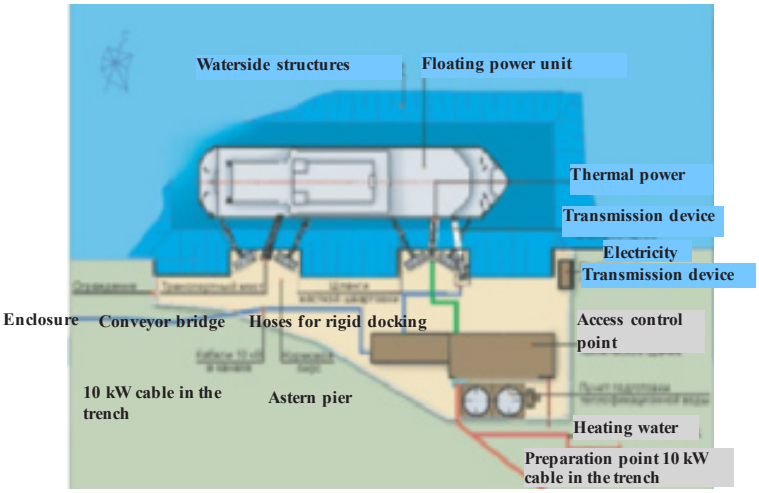


Figure 4: FCNPP Arrangement (Reproduced courtesy of OKBM Afrikantov)

Reactor Building

The general cross-section view of the FPU is given in *Figure 5*. The FPU is a flush deck non-self-propelled rack-mounted vessel with hull and multi-layer deckhouse. The medium portion of the FPU has a reactor compartment and nuclear fuel handling compartment. A turbogenerator compartment and electro-technical compartment are arranged in the ship's head with respect to the reactor compartment, auxiliary installations compartment and accommodations are arranged in astern. Each reactor plant is arranged within steel pressure containment, which is a reinforced structure of the FPU casing. The containment is designed for maximum pressure, which can develop during accidents. On-board the FPU, storages for spent cores and means are arranged that ensure reactor reloading.

Control Building

The KLT-40S reactor is controlled using the operator's automated workstation through respective control panel located in the central control room. In case it is impossible to carry out control from the central control room, information on the reactor status is obtained and safety systems are activated to make reactors subcritical and control reactor plant cooling using emergency cooling control panels located outside the central control room.

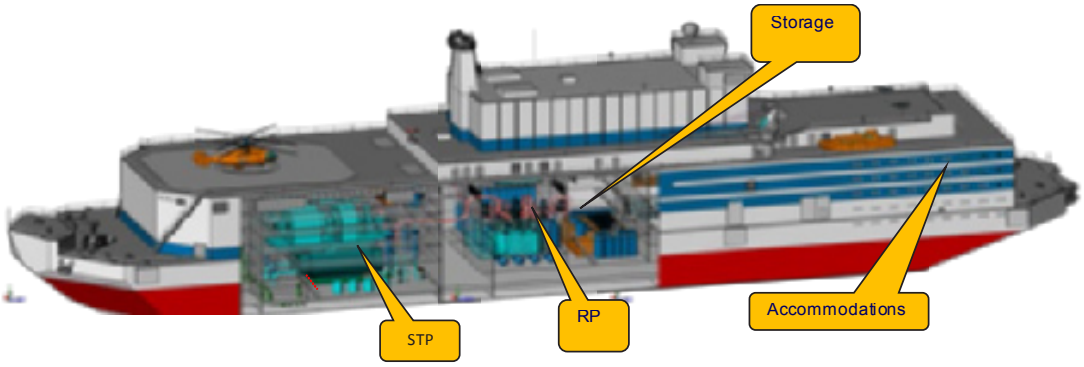


Figure 5: General cross-section view of the FPU (Reproduced courtesy of OKBM Afrikantov)

Turbine Generator Building

The steam turbine plant (STP) is intended to convert the thermal power from steam obtained in the KLT-40S reactor to the electric and thermal one to heat water in the intermediate circuit of the cogeneration heating system. The FNPP structure includes two steam turbine plants. Each STP is independent of the other and is connected to its own module of KLT-40S. Heat is delivered to the shore by heating intermediate circuit water, which circulates between FPU and the shore, using steam from adjustable turbine steam extraction.

9. Design and Licensing Status

KLT-40S is the closest to commercialization of all available FNPP designs, and expects deployment through the Akademik Lomonosov FNPP. Overall, the KLT-40S is a modified version of the commercial KLT-40-type propulsion plants employed by the Russian icebreaker fleet. 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 FNPP 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 in 2018 – 2019.

10. Plant Economics

It is expected that the economic competitiveness of the KLT-40S reactor units would be determined by the following advantages:

- FNPPs can be serially manufactured in shipyards and delivered to a customer fully assembled and ready for operation;
- The reactor is based on a well-proven design; the layout of the reactor is compact and modular, reduced amount of capital needed to prepare FNPP's docking area as compared to a site for a land-based NPP;
- Reduction in construction period, and operating costs due to need for less on-site employees.

Specific capital costs for the FPU are estimated as 5,300 $/kW(e) according to rate of 2016. Coastal and waterside structures of the FPU depend on the site location. The production cost of electricity generated by the KLT-40S FPU reactor plant is estimated as 8.8 cent/kW according to rate of Quarter I, 2016 at a discount rate of 7% (less than 10 cent/kWh).

RITM-200 (OKBM Afrikantov, Russian Federation)

1. Introduction

The RITM-200 is an integral pressurised water reactor (PWR) for multipurpose nuclear icebreaker, floating or land-based NPP producing an electrical output of 50 MW, designed by the OKBM Afrikantov. Initially, the RITM-200 was developed specifically for nuclear icebreaker ships powered by two reactor modules. Later on, it was designed for non-propelled floating nuclear power plant (FNPP). The RITM-200 can be further modified to produce heat for desalination plants and various industrial applications, e.g, marine drilling platforms. The new generation icebreaker ship using RITM-200 is designed with double-hull and an increased icebreaking capability. The RITM-200 will basically replace the KLT-40 series for both nuclear icebreaker ship and land based power plant. RITM-200 will require refuelling every 7 years in its 40-year design life.

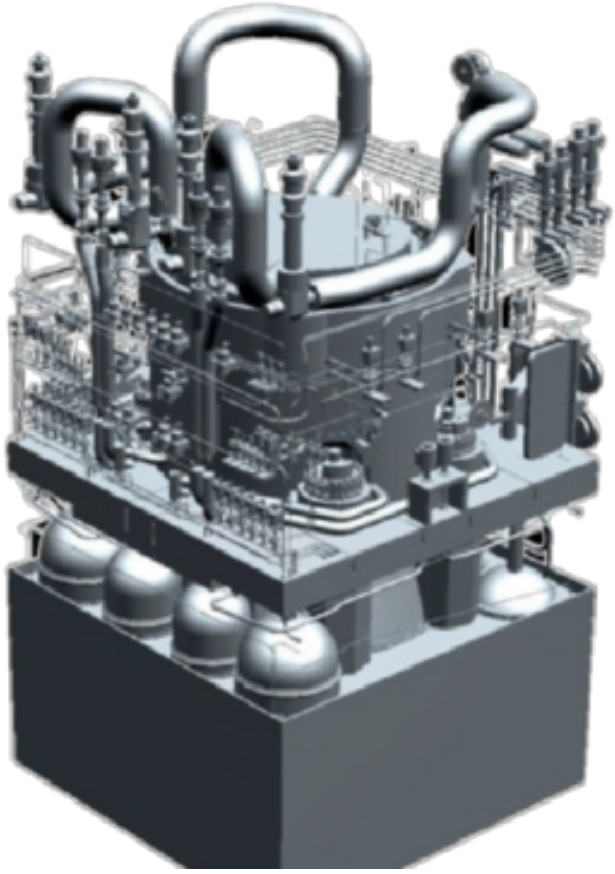


Figure 1: Reactor System Configuration of RITM-200
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2. Target Application

The RITM-200 is designed to provide the shaft power on a nuclear icebreaker. The reactor can also be used for FNPP for cogenerating heat and electricity, power and desalination complexes, and offshore drilling rigs.

3. Development Milestones

2009	Detailed design of the reactor plant finished
2012	The detailed design documentation developed for the main equipment, i.e. reactor, circulation pump. CRDMs, primary circuit filter, heat removal system HX, safety devices and valves. Fabrication of components and equipment started.
2016	Complete RPV delivery
2017	The first RITM-200 icebreaker will be in operation
2020	Two serial RITM-200 icebreakers will be commissioned

4. General Design Description

Design Philosophy

An integrated approach was adopted to determine the main parameters of the primary system, selection of equipment and layout, determining the optimal inventory and parameters of the safety systems for the RITM-200. Inherent safety characteristic is ensured by high thermal storage capacity, natural circulation capacity to cool the primary system, primary system pipelines with minimum length, nozzles equipped with leak stoppers, and large volume of primary coolant for enhanced time margin the case of loss of coolant accidents. The reduction of neutron fluence in the vessel enables the increase in radiation margin.

Nuclear Steam Supply System

The RITM-200 has an integrated nuclear steam supply system (NSSS) in which 4 steam generators are installed inside the RPV.

Reactor Core

The RITM-200 adopts a low enriched cassette core similar to that of the KLT-40S that ensures long time operation without refuelling and meets international non-proliferation requirements. The core houses 199 fuel assemblies with fuel enrichment up to 20%. The design also gives lower fluence on the reactor vessel. The core has the assigned service life of 7.0 TWh that enables it to achieve 65% capacity factor.

Reactivity Control

Control rod drive mechanism (CRDM) is used for reactivity control. A bank of emergency protection (EP) drive mechanisms is designed for fast reactor shut down and to maintain it in the subcritical condition in accidents. A bank of compensation group (CG) drive mechanisms is intended to compensate for the excessive reactivity at start up, power operation and reactor trip. The CRDM of the RITM-200 is made based on the drives used in the KLT-40S.

Reactor Coolant System

The reactor is designed as an integral vessel with the main circulation pumps located in separate external hydraulic chambers with side horizontal sockets for steam generator (SG) cassette nozzles. Each of the four SGs have 3 rectangular cassettes, while the four main circulation pumps are installed in the colds leg of the primary circulation path and separated into four independent loops. The SGs produce steam of 295°C at 3.82 MPa flowing at 248 t/h. The conventional RCPs are used. The pump is vane, single step, and has a sealed asynchronous electric motor. The motor has one winding. The rotation frequency is adjusted using conversion of the power current frequency.

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
Continuous operation period (hr)	26000
Design life (years)	40
Plant footprint (m^2)	N/A
Coolant/moderator	Light water
Primary circulation	Forced circulation by pumps
System pressure(MPa)	15.7
Core inlet/exit temperatures (^oC)	277 / 313
Core flow rate (ton/hr)	3,250
Steam generating capacity (ton/hr)	248
Steam temperature (^oC)	295
Pressure (MPa abs)	3.82
Neutron fluence at the end of life time (neutron/cm^2)	5.2×10^{19}
Main reactivity control mechanism	Control Rod Drive Mechanism (CRDM)
RPV height (m)	8.5
RPV diameter (m)	3.3
Module weight (metric ton)	1100
Configuration of reactor coolant system	Integral PWR
Power conversion process	Indirect Rankine cycle
Fuel type/assembly array	UO_2 pellet/hexagonal
Fuel assembly 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
Cogeneration capability	Yes
Approach to engineered safety systems	Combined active and passive
Number of safety trains	2
Refuelling outage (days)	40
Distinguishing features	Developed for universal nuclear ice breakers
Modules per plant	2
Target construction duration (months)	48
Seismic design (g)	3g
Core damage frequency (per reactor-year)	0.9E-6
Design Status	Components under construction, ready for installation

Reactor Pressure Vessel and Internals
The reactor pressure vessel (RPV) houses the reactor core and internals and four (4) steam generators. Four (4) vertical reactor coolant pumps are directly connected onto the RPV using pipe stubs.

Steam Generator
The RITM-200 uses once through (straight tube) SGs, of which the specific steam generating capacity is more than two times higher than the capacity of the common coil-type SGs. The configuration of the steam generating cassettes makes possible to compactly install them in the RPV.

Pressurizer
The design adopts pressure compensation gas system proved comprehensively in the Russian ship power engineering. It is characterized by a simple design, which increases reliability, compactness, and no electric power required. The compensation system is divided into two parallel independent groups to reduce the restrictor diameter in the compensatory nozzles of the steam generating unit and to decrease a coolant leak rate in primary-pipe-break accidents. It makes possible to use one of pressurizers as a hydraulic accumulator, increasing reactor plant reliability considerably in potential loss-of-coolant accidents.

5. Safety Features

The safety concept of the RITM-200 is based on the defence-in-depth principle combined with the plant self-protection and use of passive systems. Properties of intrinsic safety are intended for automatic control of power density and reactor auto-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 possible disturbances. RITM-200 optimally combines passive and active safety systems to cope with abnormal operating occurrences and design basis accidents.

- Passive pressure reduction and cooling down systems have been included (system reliability is confirmed by bench testing);
- Pressure compensation system is divided into two independent groups to minimize size of potential coolant leak;
- Main circulation path of the primary circuit is located in a single vessel;
- Steam header of primary coolant circulation is added, which ensures safety of the plant during SG and RCP failures.

The exposure dose for the crew in normal operation and design basis accidents does not exceed 0.01% of the natural radiation limit. The public exposure dose in beyond DBA with core severely damaged, does not exceed the values which require protective measures. Seawater activity due to operation is equal to 0.1 Bq/l, which is 100 times less than the specified value for drinking water.

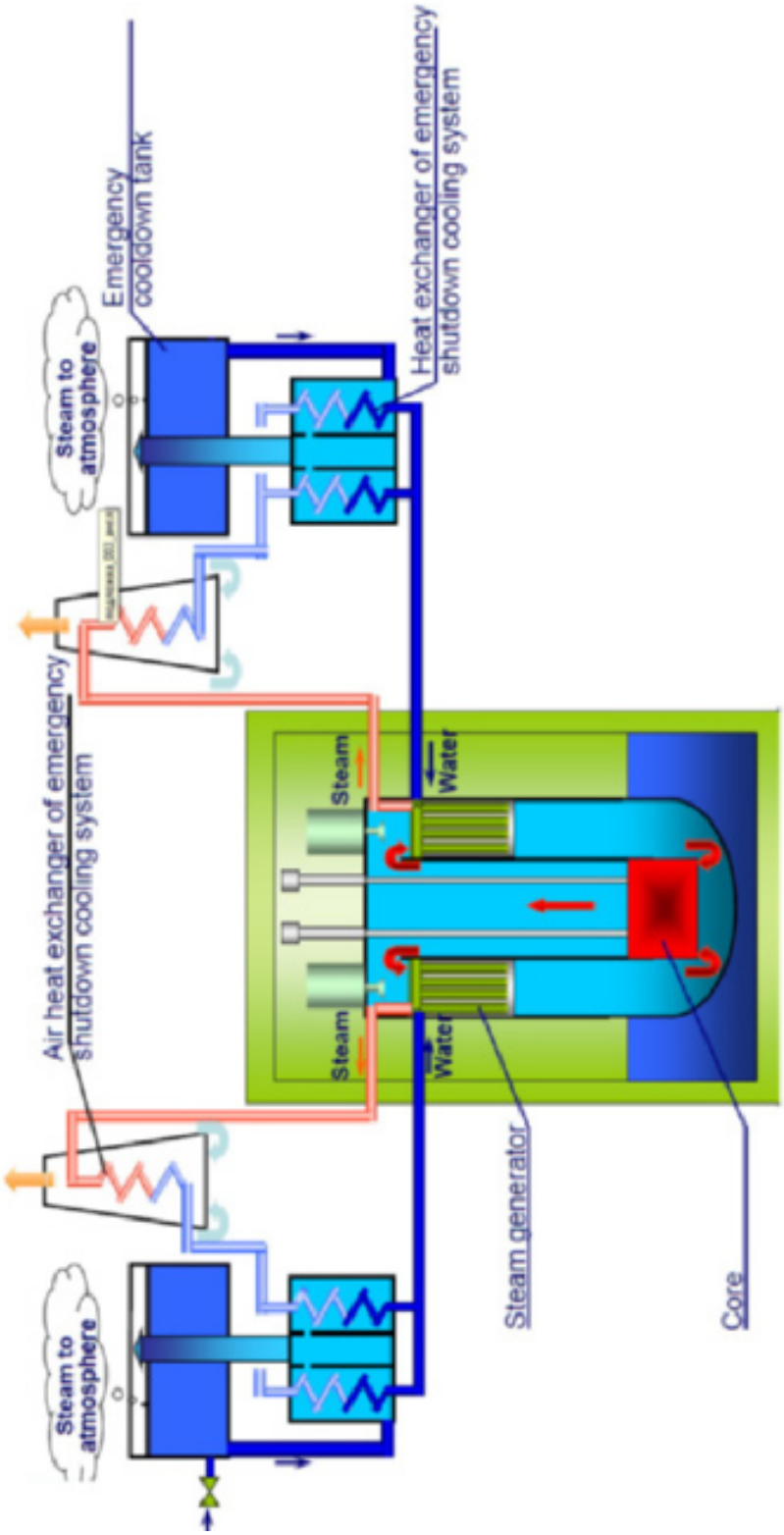


Figure 2: Emergency Core Cooling System of RITM-200
(Reproduced courtesy of OKBM Afrikantov)

Engineered Safety System Approach and Configuration

The RITM-200 high safety level is achieved both by inherent self-protection properties and a combination of passive and active safety systems. Moreover, redundancy of safety system equipment and channels and their functional and/or physical separation are provided to ensure high reliability indices. Safety systems are driven automatically by the control system, when controlled parameters achieve appropriate set points. In case of automated systems failure, self-actuating devices will actuate directly under the primary circuit's medium pressure to ensure reactor trip and initiate the safety systems. The CRDM is electrically driven that will insert emergency control rods into the core in case of station black out (SBO). The speed of safety rods in the case of emergency when driven by electric motor is 2 mm/s. The average speed of safety rods if driven by gravity is 30 to 130 mm/s.

Decay heat removal system

The decay heat removal system (DHRS) combines several independent channels connected to different SGs. Each channel contains a DHRS tank; a heat exchanger located in the DHRS tank; a water storage tank, and air cooled heat exchanger. The system uses two types of heat exchangers (HXs): water-cooled HX similar to those used in KLT-40S and air-cooled HX. At the initial stage, heat is removed by the two HXs in which the air goes directly to the atmosphere and the water goes to the DHRS tank to ensure the desired optimum heat removal. After the water in the DHRS tank evaporates, the air-cooled HX would still operate and provide cooling to the core for unlimited period. This combination enables a minimized overall dimension of the HXs and water storage tank.

Emergency core cooling system

The Emergency core cooling system (ECCS) is based on active and passive operating principles, a two-channel design with redundancy of active elements in each channel. Each channel comprises hydraulic accumulators supplying water to the SG passively by gas pressure, and makeup pumps supplying water to the SG when power supply is available.one group of pressurizers is used as hydraulic accumulator when the other group of pressurizers is depressurized.

Containment system

The RITM-200 module is installed inside containment with a volume of 6m×6m×15.5m. The integrated NSSS configuration significantly reduces the reactor dimension inside the containment. The mass of the containment with reactor module is 1100 ton. The containment pressure reduction system is passive-driven and provides localization of steam-water mixture during LOCA in the containment of the damaged RITM-200 module, by using self-actuating devices like hydraulically operated pneumatic valves (at increase of pressure in containment) and pressure-actuated contact breakers.

6. Plant safety and Operational Performances

The assigned service life of the plant for replaceable equipment is 20 years with a continuous operation period of 26000 h.

7. Instrumentation and Control systems

An automated control system is provided in the RITM-200 nuclear power plant to monitor and control plant processes. This system possesses necessary redundancy with regard to safety function fulfilment and allows both automated and remote (operator's) control of the power plant.

8. Plant Arrangement

The basic architectural design for the land-based power unit is to place the reactor , including its servicing systems, spent fuel pool, and auxiliaries, in a double protective containment resistant to the air-crash. The designers also claim that the overall size of the module allows transport of the reactor by rail.

Reactor building

RITM-200 plant with service equipment is arranged in a building providing protection against external impacts. The building structure is also provided with biological shielding to limit impact on the environment and population during reactor operation.

Control building

Station operation is controlled from the main control room by plant operators . Information control and representation means possess necessary redundancy to provide station operation monitoring and control in emergency and failure conditions.

Balance of plant:

The RITM-200 design can be configured for land-based stationary applications wherein the reactor system – nuclear island coupled to a turbine island and ancillary buildings for spent fuel storage, water treatment, maintenance, and switchyard connections with configurations similar to conventional large LWRs, are housed within a relatively small footprint.

Turbine Generator building

Each RITM-200 reactor can be thermally coupled with one or multiple turbine-generator sets. A slightly superheated steam is supplied to the turbine in the secondary circuit with part of the steam taken off from the turbine and directed to the heat exchanger of a district heating circuit.

Electric power systems

The electric power system consists of power generation system and power distribution system. The structure of this system depends on a specific station location and Customer's needs. The house load power supply system for station consumers is also incorporated which provides power supply for station consumers under normal and emergency conditions.

Floating nuclear station based on the RITM-200 reactor plant

The RITM-200 reactor plant also could be a base to create a floating nuclear power station. Moreover, it is possible to operate this station without refueling at its site. This fact will make it more commercially attractive, and availability of near term prototype – the power plant of the universal nuclear icebreaker – will reduce time and cost of this plant.

9. Design and Licensing Status

The RITM-200 design was developed in conformity with Russian laws, norms and rules for marine nuclear power plants and safety principles developed by the world community and IAEA recommendations. An order for the first commercial FNPP with RITM-200 is expected to be placed in the near future, while the first icebreaker adopting this technology is expected to be commissioned in the next few years. The RITM-200 adopts optimal combination of passive and active safety systems. The components for the two modules of RITM-200 are being manufactured for the first multipurpose icebreaker targeting potential complete delivery in 2016. Serial deliveries of reactor modules for two consequent nuclear ice-breakers

will be in 2017 and 2018. RITM-200 has been chosen for the next generation of Russian Federation ice-breakers and accordingly costs for the design will decrease as reactors are built and installed.

10. Plant Economics

The average specific cost of construction for the land-based station with the RITM-200 plant will be approximately 6000$/kW, while the electricity unit cost is approximately 0.095$/kW·h. The average specific cost of construction for the floating station with the RITM-200 plant will be approximately 3900$/ kW, while the electricity unit cost is approximately 0.062$/ kW·h. The shown cost characteristics do not consider coastal facilities and infrastructure of the station, because they depend considerably on a site (geophysical data and climatological conditions, valid local regulatory documents, social economic and infrastructure conditions, peculiarities of a selected operation model and other factors).

VBER-300 (OKBM Afrikantov, Russian Federation)

1. Introduction

The VBER-300 is a multipurpose medium-sized power reactor with a rated electric power of 325 MW intended for land-based nuclear power plants (NPPs), nuclear cogeneration plants, and transportable floating nuclear power plants (FNPPs). The VBER-300 design is evolution of modular marine propulsion reactors. An increase in thermal power causes an increase in mass and overall dimensions; however, the reactor basic design is similar to that of marine propulsion reactors. The VBER-300 design was developed based on the lessons learned from the design, safety and operating experience for VVER reactors. VBER-300 adopts proven nuclear ship building technologies and operating experience that in turn contribute to enhancement of operational safety and reduction in production costs. VBER-300 can be configured as a multi-module plant on request of the customer. VBER-300 design features are availability for both land-based and transportable FNPPs, a variety of cogeneration options, maximally compact design, improved plant efficiency, and protection against external impacts. A reduction in construction time is achieved due to the compact design of the reactor system.

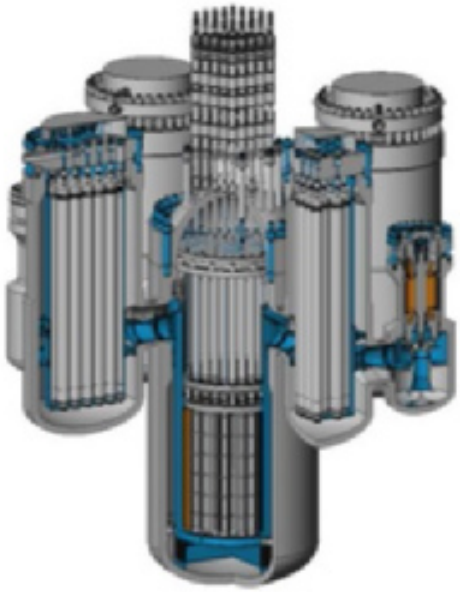


Figure 1: VBER-300 Reactor System Configuration
(Reproduced courtesy of Afrikantov OKBM)

2. Target Application

The VBER-300 nuclear plants are intended to supply thermal and electric power to remote areas where centralized power is unavailable, and to substitute capacities of available cogeneration plants on fossil fuels. The design is also proposed to be used as a power source for seawater desalination complexes. The VBER-300 nuclear plant has two reactor units that operate in the steam-condensing mode and can generate 600 MW(e) to satisfy power demands of a city with a population of 300000. According to the OKBM's data, when VBER-300 has cogeneration capabilities, the total electric output will reduce to 200 MW(e) providing 460 Gcal/hr for process heat applications.

3. Development Milestones

2001	Design activities to develop VBER reactors started
2002	Technical and commercial proposal for the two-unit VBER NPP
2004	preliminary design 1 approved by the Scientific and Technical Board and State Nuclear Supervision Body (GosAtomNadzor)
2006	JSC "Kazakhstan-Russian company "Atomic stations" was established to promote the VBER-300 design.
2007–09	Technical Assignment for the NPP design and final designs of the reactor plant, automated process control system, and heat-generating plant; feasibility, economy, and investment studies of the VBER-300 RP NPP for the Mangistau Region, Kazakhstan
2007–08	Development of the 100–600 MW VBER plant
2008–11	R&D for the VBER-460/600 NPP design
2011–12	Development of the VBER-600/4 NPP based on the heat exchange loop of the increased capacity
2012–2015	Technical and economic optimization of the VBER-600/4 plant Development of the conceptual design for the two-unit VBER-600/4 NPP

4. General Design Description

VBER-300 technical characteristics are modular design, once-through steam generators with titanium tubes, canned main circulation pumps, low density core, and optimal combination of active and passive safety systems. VBER-300 has some specific features, such as a double containment with low possibility of radioactive release under all accident scenarios, an ultimate heat sink by air cooling at minimum requirement for water cooling, a reduced nuclear steam supply system (NSSS), a simplified balance of plant, and suitability for both land-based stations and transportable FNPPs.

Design Philosophy

VBER-300 design using ship-based modular configuration enhances the safety philosophy through proven marine modular technologies. The reactor design has no pipelines in the primary circulation circuit. All vessels of main equipment such as reactor vessel, steam generators , and main coolant pump(MCP) are welded to each other to make a single reactor unit as shown in *Figure.1*. The VBER-300 reactor unit incorporates the reactor and four steam generator – MCP two-vessel units. The compact VBER-300 reactor system comprises the steam generating system in a limited space of the reactor compartment, and has enhanced reliability and long refuelling cycle. VBER-300 can also be configured as a transportable FNPP and can be arranged to operate individually or as multi-module plant, increasing the power output by means of scaling up the equipment and with the same reactor system configuration.

Reactivity Control

In the VBER-300 reactor, 61 control rods in combination with fuel elements mixed with burnable poison materials provide safe and reliable reactivity control during both normal and transient operating period. Control rods are operated through high-performance electromechanical control rod drive mechanisms (CRDMs). The control rods elements are designed to maintain the core subcritical even if the most reactive assembly fails (i.e. stuck-rod/assembly event). To compensate for the fuel burnup reactivity margin, fuel rods with gadolinium burnable poison contained in uranium dioxide pellets are distributed across each FA with configurations similar to those used in VVER-1000 reactors. Boric acid is also dissolved and maintained at controlled concentrations within the primary coolant system to ensure optimum core power distribution.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	Afrikantov OKBM
Country of origin	Russian Federation
Reactor type	Integrated PWR
Electric power (MW(e))	325
Thermal power (MW(th))	917
Expected capacity factor (%)	92
Design life (years)	60
Plant area (ha)	30 (two-unit nuclear cogeneration plant)
Coolant/moderator	Light water
Primary circulation	Forced circulation
System pressure (MPa)	16.3
Core inlet/exit temperatures (^{o}C)	292 / 328
Main reactivity control mechanism	Control Rod Drive Mechanism and soluble boron
RPV height (m)	9.3
RPV diameter (m)	3.9
RPV weight (metric ton)	200
Height of the reactor unit, m	15.3
Circumscribed diameter of the reactor unit, m	11.2
Mass of the reactor unit (w/t water), t	1,435
Configuration of reactor coolant system	Integrated
Power conversion process	Indirect Rankine Cycle
Fuel type/assembly array	UO_2 pellet/hexagonal
Fuel assembly active length (m)	3.5
Number of fuel assemblies	85
Fuel enrichment (%)	4.95
Fuel burnup (GWd/ton)	50
Fuel cycle (months)	72
Cogeneration capability	Yes
Approach to engineered safety systems	Hybrid (active and PASSIVE) system
Number of safety trains	2
Refuelling outage (days)	Maximum 24 days to refuel the core and carry out periodic maintenance Maximum 16 days only to refuel the core
Specific features	Power source for transportable FNPPs, a variety of cogeneration
Modules per plant	1
Target construction duration (months)	40
Seismic design (g)	0.25
Core damage frequency (per reactor-year)	1E–6
Design status	Licensing stage

Reactor Pressure Vessel and Internals

The reactor pressure vessel (RPV) consists of the reactor core and internals with an overall height of 9.3m and a diameter of 3.9m. The VBER-300 design provides a special system of emergency vessel cooling to solve the problem of retaining the melt inside the reactor vessel in severe accidents. The core melt retention is facilitated by the low power density, relatively low level of residual heat, no penetrations in the reactor vessel bottom and smooth outer surfaces of the reactor vessel bottom creating more favourable conditions for steam evacuation under core cooling by boiling water.

Reactor Coolant System

The VBER-300 primary cooling mechanism under normal operating conditions operates using forced circulation of coolant by the MCPs and using natural circulation in the shutdown condition. The reliability and operational safety of the MCPs are enhanced due to the usage of a proven technology and operating experience for the pumps in the area of marine propulsion. The MCPs are connected directly to the steam generators (SGs). All components of the primary loop are directly connected to the RPV, except for the pressurizer. The MCPs are centrifugal single-stage canned pumps with impellors.

Steam Generator

The SGs are once-through coil modules with the secondary coolant flowing inside the tubes. The feedwater is pumped through an inlet in the SG head, circulates within the SG tubes and exits through the SG outlet as a superheated steam at the design pressure and temperature for expansion in the turbine generator units.

Pressurizer

The VBER-300 has an external steam pressurizer that is conventional for loop PWRs. The water region in the pressurizer, where electric heaters are located, is connected with the SG hot section in one primary loop. The steam region of the pressurizer is connected with the cold section in this loop near the MCP pressure chamber, from which the underheated water is supplied to the pressurizer when valves are open in the injection line. The pressurizer head in the steam region has two safety valves that protect the primary circuit against overpressure in case of accidents with loss of decay heat removal.

5. Safety Features

The VBER-300 safety systems are based on the defence-in-depth principle with redundancy relying on passively driven systems that enables the core to operate within safety margins under all anticipated accident scenarios for at least 24 hours. After this initial period, emergency back-up and diverse safety systems ensure continued core cooling for extended time. In addition, separation of the passive and active cooling channels prevents common failures of the emergency core cooling systems (ECCS).

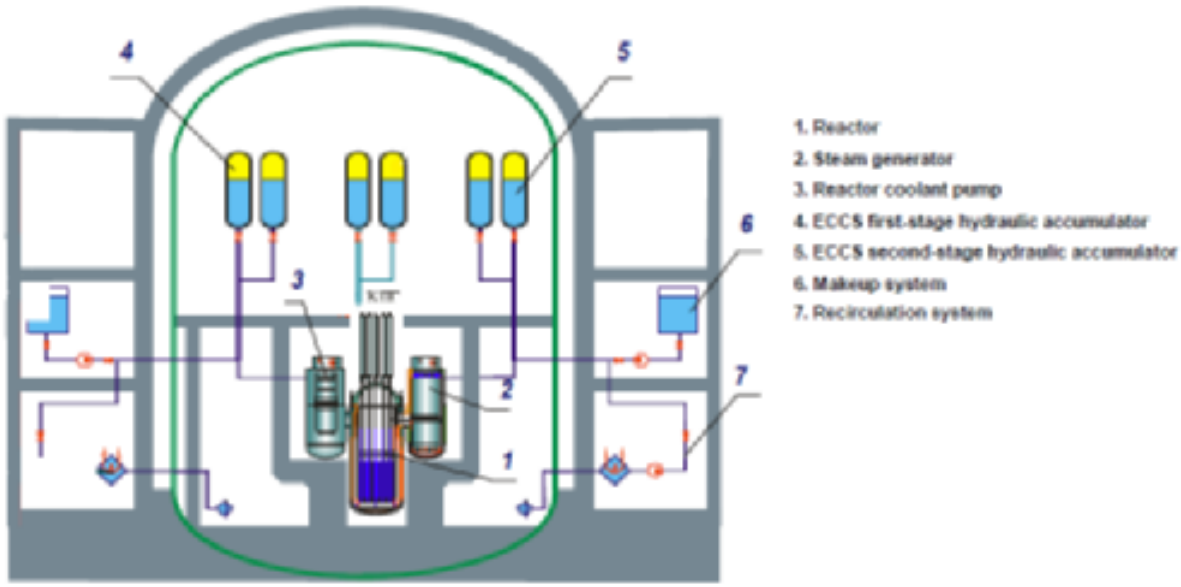


Figure 2: VBER-300 Emergency Core Cooling System
(Reproduced courtesy of Afrikantov OKBM)

Engineered Safety System Approach and Configuration

The safety assurance and engineering solutions incorporated in the design focus on accident prevention measures, design simplification, inherent safety; passive safety systems and enhancement of safety against external events (including acts of terrorism); and mitigation of severe accident consequences. The RPV and connecting piping that usually form the primary pressure boundary represent an additional physical barrier. The leak-tight carbon steel containment and protective enclosure with filtration forms the ultimate barrier separating the reactor system from the environment. For all cogeneration applications, the separation of heat exchangers represents a physical barrier to prevent radioactive release. For the VBER-300 district heating configuration as shown in *Figure.* 3, a physical barrier is formed by the independent thermal-hydraulic loop comprising the district heating pump, associated piping, and radiators.

Decay Heat Removal System

The decay heat removal system (DHRS) consists of two passive heat exchangers and a process condenser. Passive safety features are intended to arrange recirculation in the core for the removal of decay heat in the course of scheduled maintenance, refuelling or under emergency conditions. Passive emergency shutdown cooling system operates using natural circulation of coolant in all heat transport circuits with stored water tanks, where water is evaporated and condensed back to liquid upon a contact with the cooler surfaces of the containment inner shell. Decay heat is also removed indirectly by the secondary circuit using the steam turbine condenser.

Emergency Core Cooling System

The ECCS contains two stages accumulators with different flow-rate characteristics to ensure emergency core cooling for 24 hours, makeup pumps and a recirculation system. If electrical power is available during accidents, makeup pumps and an active recirculation system ensures emergency core cooling beyond the initial 24 hours. The accumulators operate in two stages. The VBER-300 emergency shutdown system consists of the control rod drive mechanisms, two trains of liquid absorber injection, and two trains of boron control from the make-up system. Emergency residual heat removal system (RHRS) by means of passive

cooling channels with water tanks and in-built heat exchangers, ensure reliable cooling up to 72 hours and longer. The system is actuated by passive means—hydraulically operated pneumatic valves. The emergency core cooling accumulators are part of the passive water injection system as injection is done using compressed gas. Containment depressurization systems prevent containment damage and reduce radioactive release in design basis accidents (DBA) and beyond DBAs. A small and medium loss of coolant accidents (LOCA) are prevented by a combination of a sprinkler system, low-pressure emergency injection system, and core passive flooding system.

Containment System

The land-based VBER-300 containment system includes a double protective pressure envelope formed by an inner carbon steel shell and an outer reinforced concrete containment structure. In addition, localizing reinforcement is provided to protect the pressure boundary represented by all auxiliary systems hydraulically connected to the primary loop. The containment is designed to withstand all stressors induced by all credible accident scenarios, including aircraft crashes. The inner steel containment of 30 m in diameter and 49 m high provides space for condensing the steam generated from the medium in large LOCAs. The outer concrete structure 44 m high and 36 m in diameter serves as protection against natural and man-caused impacts.

6. Plant Safety and Operational Performances

The VBER-300 safety concept is based on the state-of-the-art defense-in-depth principles aimed at prevention of accidents and mitigation of accident consequences, including consequences of severe accidents, mainly due to well-developed self-protection properties of the modular PWR and wide application of passive and active safety systems. With the modular configuration, the VBER-300 has increased resistance to impact loads in case of an earthquake and an aircraft crash. Results of strength analysis under seismic loads of up to 8 points according to the MSK-64 scale carried out for the VBER-300 RPs confirmed that the VBER-300 reactor unit has a two-times safety margin (maximum seismic stress of the most loaded vessel unit is 150 MPa maximum at an allowed stress of 370 MPa). Analysis of the 20 t aircraft crash on the VBER-300 reactor compartment showed that overload upon the attachment fittings of the reactor unit is less than seismic loads. It is considered that core melting accidents for the VBER-300 core are postulated. In the course of the severe accident, the reactor cavity is filled with water from the emergency reactor vessel cooling system ensuring reliable heat removal from the external surface of the bottom and lower portion of the vessel. Retention of satisfactory mechanical properties and load-carrying capacity of the vessel ensures retention of the melted core inside the reactor. The safety level of the power units with VBER plants correspond to requirements for Generation III+ advanced nuclear stations making it possible to place them near cities that is of extreme importance as virtually all regional power sources are used for district heating. The buffer area of the station coincides with the perimeter of the industrial site. The calculated radius of protection measures planning for population is 1 km.

7. Plant Arrangement

In the basic architecture of the land-based VBER-300 power unit, the reactor, including its servicing systems, spent fuel pool, and auxiliary equipment are arranged within a double containment resistant to aircraft crashes.

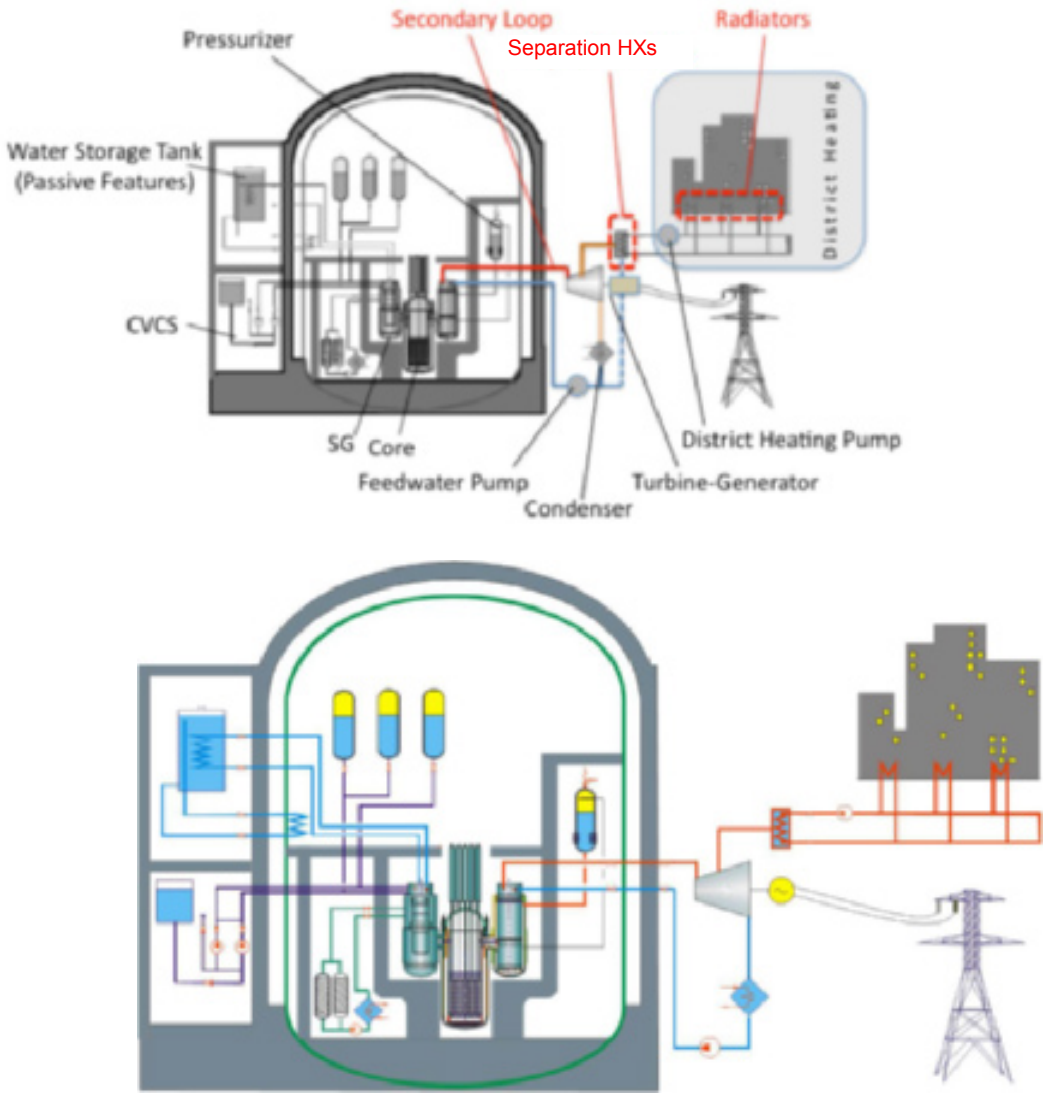


Figure 3: VBER-300 electric and district heating application
(Reproduced courtesy of Afrikantov OKBM)

VBER-300 as a Transportable Floating Nuclear Power Plant

VBER-300 is a non-propelled transportable FNPP designed according to the Russia's Sea Navigation Register. The transportable FNPP is located on a platform consisting of three pontoons (one central and two peripheral ones). The independent units are in-line located on the central pontoon. Each unit consists of a reactor compartment, power plant control board and electrical equipment compartment; 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. Fuel storage is located on the central pontoon between the units. 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 (up to 220 kV) to coastal facilities and that for power supply to the plant in-house loads are located on the left board pontoon. A composite steel-iron concrete vessel of the power unit eliminates the need for scheduled docking during the plant lifetime. A period between overhauls (20 years) is defined by the key reactor plant equipment life that is about 15000 h. The VBER-300 FNPP operating life is 60 years.

Reactor Building
The inner steel shell is a leak-tight cylindrical enclosure 30 m in diameter that is covered with the semispherical dome 15.0 m in radius and that has an elliptical bottom. The height of the leak-tight enclosure is 47.0 m. The steel shell is designed for parameters of the maximum DBA with the excess pressure of 0.4 MPa and the temperature of 150 C. The outer protective enclosure is made of one-piece reinforced concrete without preliminary tensioning of the steel and consists of a cylindrical portion of the semispherical dome. Building structures of the outer protective enclosure are designed for external accidental exposures, including an aircraft crash and air shock wave.

Balance of Plant:
The VBER-300 design can be configured for land-based stationary applications wherein the reactor system—a nuclear island coupled to a turbine island and auxiliary buildings for spent fuel storage, water treatment, maintenance, and switchyard connections with configurations similar to conventional large LWRs—is housed in a relatively small area.

Turbine Generator Building
Each VBER-300 reactor system can be thermally coupled with one or multiple turbine generator sets. A slightly superheated steam is supplied to the turbine in the secondary circuit with part of the steam taken off from the turbine and directed to the heat exchanger of a district heating circuit. A VBER-300 nuclear station can operate as a NPP with a condensing turbine and as a nuclear cogeneration plant with a cogeneration turbine.

8. Design and Licensing Status

Development of the final design and design documentation for a VBER-300 nuclear station can begin immediately upon the request of a customer. It will take 36 months to develop documentation to the extent needed to obtain a license for VBER-300 NPP construction, including 18 months to develop the technical design.

9. Plant Economics

Calculations of construction costs and electric power generation costs performed in 2006–2009 for VBER-300 nuclear units showed that these can compete with carbon electric power sources for isolated regions that are not included into large power systems and for regions that strongly depend on the cost of fossil fuel purchase and transportation.

ABV-6E (OKBM Afrikantov, Russian Federation)

1. Introduction

The ABV-6E is reactor plant (RP) as a part of nuclear power system (NPS) that produces 14 MW(th) and 6 MW(e) in cogeneration mode or 9 MW(e) in condensation mode. ABV-6E integral PWR adopts natural circulation of the primary coolant. The ABV-6E design was developed using the operating experience of WWER reactors and recent achievements in the field of nuclear power plant (NPP) safety. The main objective of the project is to develop small, shop fabricated, multipurpose transportable NPP for 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 will be carried out at dedicated facilities.

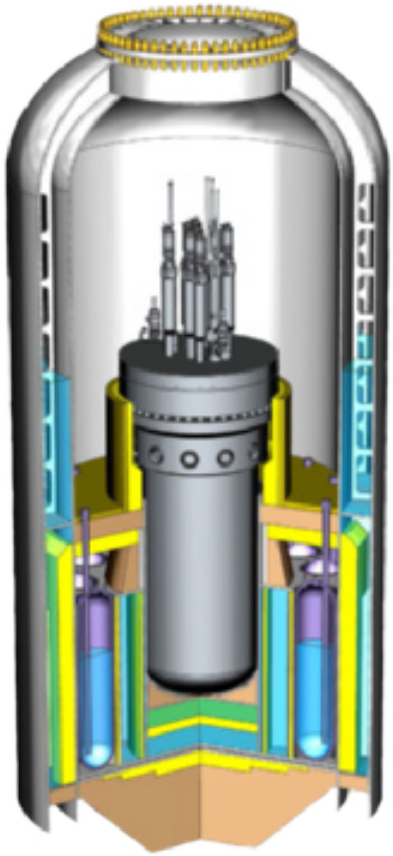


Figure 1: Reactor System Configuration of ABV-6E
(Reproduced courtesy of OKBM Afrikantov)

2. Target Application

The ABV-6E RP is intended as a multi-purpose RP. The RP is designed with the capability of powering a floating power unit (FPU) as a part of floating nuclear power plant (FNPP) with a maximum length of 91.6 m, a beam of 26 m, a draft of 3.6 m and a displacement of 8100 t. Depending on the needs of the region, the FNPP 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.

3. Development Milestones

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

4. General Design Description

The basic design approach for the ABV-6E is optimization of the capital and operational costs. The ABV-6E adopts design methodologies for minimizing mass and overall dimensions, feasibility to deliver Reactor Pressure Vessel (RPV) by railway or water transport with minimal installation activities at the construction site. 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 FPUs it provides significant reduction of displacement and feasibility to locate them in the sea shallow water areas and in the rivers. The ABV-6E is particularly applicable for modular-transportable power unit (MTPU) in sea shallow water areas. The land based MTPU with ABV-6E is fabricated as large, ready-made modules. 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 would be 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. Structural optimization of main equipment allows for delivery of the power units at the base site through water with minimum technological issues.

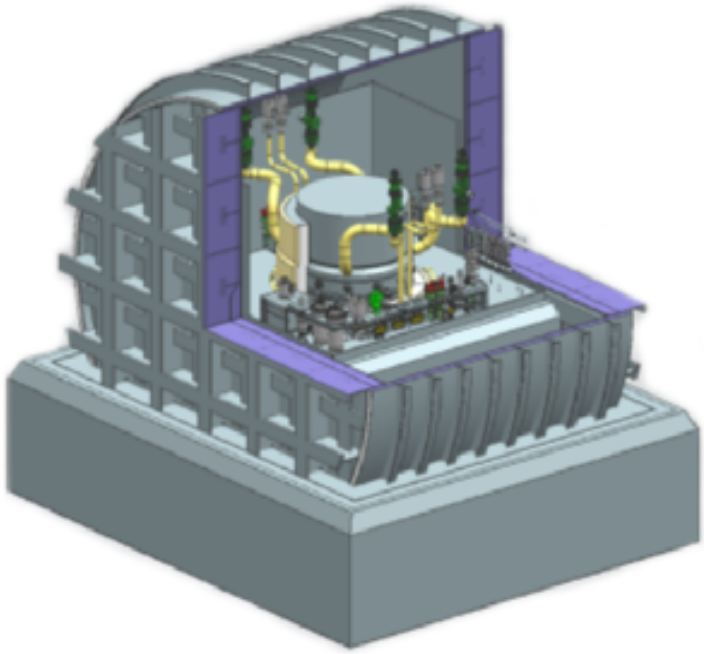


Figure 2: The conceptual view of the MTPU with ABV-6E
(Reproduced courtesy of OKBM Afrikantov)

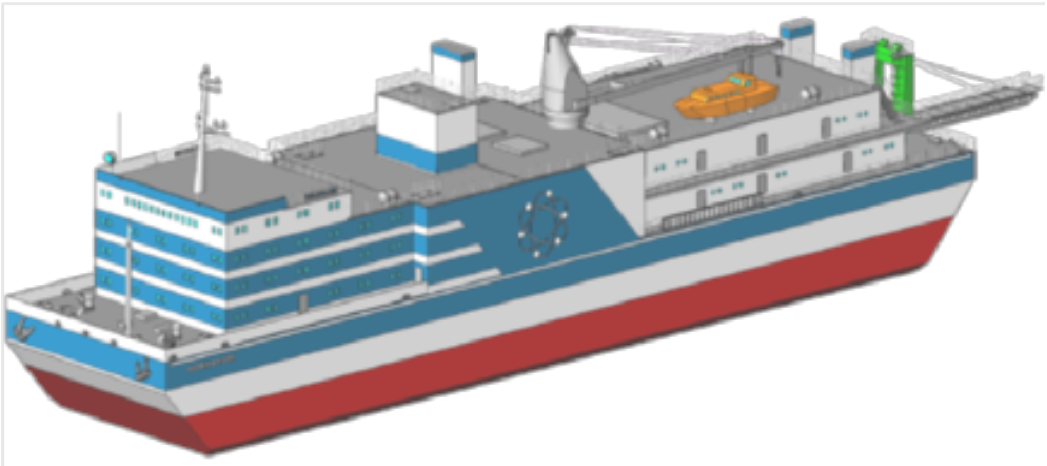


Figure 3: The conceptual view of the FPU with ABV-6E
(Reproduced courtesy of OKBM Afrikantov)

Design Philosophy
The ABV-6E is a pressurized water reactor (PWR); its design incorporates the following main features:

- Integral primary circuit layout with natural circulation of the primary coolant;
- Negative feedbacks and enhanced thermal inertia;
- Passive and self-actuated safety systems;
- Increased resistance to extreme external events and personnel errors;
- Use of nuclear fuel with the enrichment of less than 20% by weight (maximum enrichment 19.7%).

Nuclear Power System
NPS consists of reactor, steam-turbine, part of electric power system and control systems. The RPV operates under conditions of 16.2 MPa in the reactor pressure vessel. The steam generators located inside the RPV generate 295°C steam at 3.83 MPa flowing at 55 t/h. The RPV head is located under biological shielding and the control rod drive mechanism is located above the shield outside the vessel.

Reactor Core
The core comprises 121 hexagonal fuel assemblies (FA) of cassette type with active part height of 900 mm, similar to the FAs in KLT-40S. Cermet fuel is used with maximum 19.7% enriched uranium-235. Special stainless steel is used as fuel cladding.

Reactivity Control
Reactivity control without boron solution in the primary coolant and compensation of reactivity changes in power operation is achieved by mechanical control and protection system (CPS). These inherent safety features ensure automatic power regulation in a steady state operation, self-limiting power rise in case of positive reactivity insertions, automatic control of the reactor power and primary coolant pressure and temperature in transients, as well as the emergency shutdown of the reactor core including the cases with a blackout and RPV flip-over (with account of the time that the vessel flip-over process takes).

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	OKBM Afrikantov
Country of origin	Russian Federation
Reactor type	PWR
NPS electrical capacity in cogeneration mode (MW(e))	6
NPS thermal capacity in cogeneration mode (MW(th))	14
NPS electrical capacity in condensation mode (MW(e))	9
RP thermal capacity (MW(th))	38
Expected capacity factor (%)	> 70*
Design life (years)	40
Plant footprint (m^2)	20000*
Coolant/moderator	Light water
Primary circulation	Natural circulation
System pressure (MPa)	16.2
Core inlet/exit temperatures (^oC)	250 /325
Main reactivity control mechanism	Control Rod Driving Mechanism (CRDM)
RPV height (m)	6
RPV diameter (m)	2.4
RPV weight (metric ton)	88.2
Configuration of reactor coolant system	Integral type
Power conversion process	Indirect Rankine cycle
Fuel type/assembly array	UO_2/hexagonal
Fuel assembly active length (m)	0.9
Number of fuel assemblies	121
Maximum fuel enrichment (%)	19.7
Fuel burnup (GWd/ton)	N/A
Fuel cycle (months)	120-144
Cogeneration capability	Yes
Approach to engineered safety systems	Passive
Number of safety trains	2
Refuelling outage (days)	Refueling is included in the integrated maintenance
Distinguishing features	Natural circulation in the primary circuit
Modules per plant	1*
Target construction duration (months)	54 - 60*
Seismic design (Richter scale points)	7*
Core damage frequency (per reactor-year)	10E–6
Design Status	Final design

* - The values are given for the basic design. If the design is adjusted for a specific site, the shown values may change.

Reactor Pressure Vessel and Internals

The RPV is a welded cylindrical "container" with an elliptical bottom. At the top of the vessel there are pipes for feedwater supply and superheated steam removal, as well as those for the connection of the primary circuit systems and the auxiliary process systems. The RPV

head consists of a load-bearing slab, a shell attached to this slab and sealed by a weld, and a top slab welded to the shell. The cavity between the top slab and the load-bearing slab is filled with serpentine which acts as a biological shielding, and the heat insulation is located at the top. The posts of the CPS drives and thermal converters, etc. are welded to the load-bearing slab and penetrate through the cover. Points of penetration through the top slab are sealed. Fuel assemblies are located in the in-vessel shaft. The protective tubes and devices provide the necessary coolant flow rate distribution between the fuel assemblies and an arrangement of connectors for joining the absorber elements of fuel assemblies into CPS control rods and connecting the CPS control rods to CPS drives.

Reactor Coolant System

Core heat removal is based on conventional two-circuit methodology. The core is cooled and moderated by water through natural circulation of coolant in the primary circuit. Hot coolant is cooled in a once-through steam generator, where slightly superheated steam is generated, then supplied to the turbine. This design eliminates large-diameter pipelines in the primary circuit and main circulating pumps. The steam generator (SG), arranged in the annular space between the vessel and the in vessel shaft, is a once-through vertical surface-type heat exchanger generating steam of the required parameters from heat of the primary circuit coolant. The SG is divided into four independent sections; feedwater supply and steam removal from each section is carried out through the pipes in the reactor vessel. Counter flow circulation is used, i.e., the primary circuit coolant moves downward in the inter-tube space, while the secondary circuit coolant is moved upward in the tubes. In case of inter-circuit leaks, it is possible to cut off any section automatically or remotely. Identification of the leaking section is carried out with the use of the detection blocks of the radiation and process control system. Finding and disabling a faulty module is carried out during reactor shutdown.

5. Safety Features

Engineered Safety System Approach and Configuration

Safety of the ABV-6E RP is of utmost importance considering its close proximity to public area and at the same time far-off location from main technical bases, which could provide timely technical support. In view of its small power the emergency systems are simpler and often do not require active systems performance. Land-based and floating power units use the advanced active and passive safety systems for emergency cooling over an unlimited time during design-basis and beyond design-basis accidents. Low thermal capacity of reactor allows using natural circulation at primary coolant circuit and passive safety systems as primary safety systems. The auto-protective features of the NPP have been improved for deployment in far flung territories.

The safety systems include:

- passive heat removal system;
- passive core cooling system;
- reactor caisson water flooding system;
- Backup liquid absorber injection system.

Decay Heat Removal System

In emergency modes, a combined-type emergency cooldown system (ECDS) is used to remove decay heat. This system functions on natural physical processes and - because there is an air heat exchanger cooled by the atmospheric air - ensures that the decay heat is being removed from the reactor for an unlimited time in all types of accidents. Because of this, and considering the measures taken to enhance the reliability of the passive ECDS, there are no active ECDS channels in the ABV 6E reactor design, which allows the output of emergency

power supply sources to be reduced. The passive ECDS is made of two independent channels connected to two SGs each. Either channel, independently of the operability of the other channel, is capable of performing the ECDS functions, i.e. of maintaining the parameters of the primary circuit in the design limits for an unlimited time.

Emergency Core Cooling System

The emergency core cooling system (ECCS) is designed to compensate for the primary coolant leak and to cool the reactor core in case of LOCA. The ECCS comprises of the high-head pumps that inject water into the RPV if power supply is available, and the hydro-accumulators that supply water under the action of the compressed gas.

Containment Systems

The metal-and-water shielding (MWS) tank is a substantial structure for the equipment of the RP. RPV, two pressurizers and the cooler of the purification and heat removal system are enclosed inside the dry caissons of the MWS tank. The passive reactor caisson water flooding system is designed to protect the RPV against melt-down in severe beyond-design-basis accidents associated with core damage. The system feeds the primary coolant condensate to the RPV caisson. It is also possible to supply water from the fresh water intake and pumping system. The structure of the reactor caisson ensures the stable heat exchange between the RPV and MWS tank.

6. Plant Safety and Operational Performances

The NPP with ABV-6E generates electricity and heat in the power range of 20–100%N_{nom} with the continuous operation time of 26000 hours. The NPP is designed for the manoeuvring rate of up to 0.1%/s. As a protection against the external events, the NPP is equipped with both ground and waterside security structures. The structures are designed for the sites in the Arctic zone with the frost penetration as deep as 2 m. The FPU and NPP design is intended to withstand the 10-ton aircraft crash. As the analysis of emergencies has shown, the radiation and ecological impact to the personnel, public and the environment during normal operation, abnormal operation, including the design-basis accidents, does not lead either to the excess of the radiation doses established for the personnel and public, or release of any of radioactive content in the environment. This impact is also limited in beyond-design-basis accidents.

7. Design and Licensing Status

The final design of ABV-6E has been accomplished. The design has not been licensed yet.

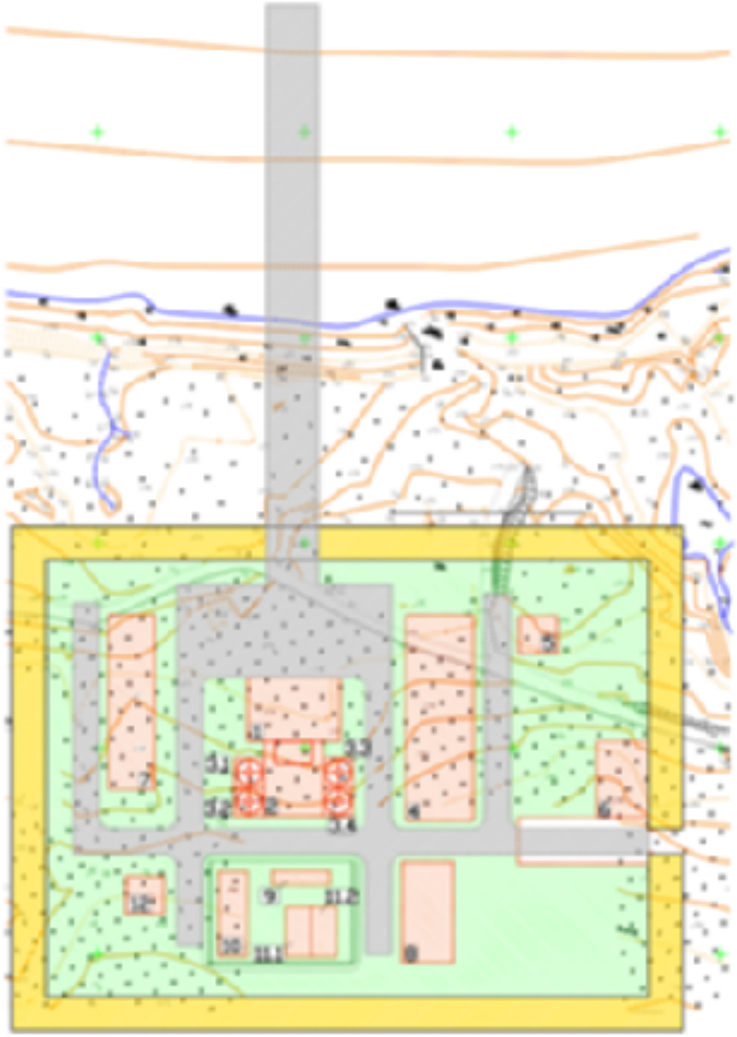
8. Plant Economics

The average specific cost of construction for the serial power unit with ABV-6E is around 8,500 $/kW. The contribution of the serial power unit to the prime cost of electricity is around 0.14 $/kW·h, The cost characteristics are shown without taking into account the costs of the coastal structures and plant infrastructure, because they depend, in a great extent, on the site (geo-physical data, climatic conditions, applicable local regulatory acts, social-economic and infrastructure conditions, features of the selected operation model and other factors).

9. Plant Arrangement

No. in plan	Title
1	Reactor plant building
2	Turbine generator module
3.1...3.4	Dry cooling towers
4	Integrated process building
5	Treatment facilities
6	Access control point
7	Guard room
8	Administrative building/shelter
9	Indoor switchgear
10	Substation-shared control room
11.1, 11.2	Diesel-generator sets
12	Garage

Figure 4: General plan of the FNPP (Reproduced courtesy of OKBM Afrikantov)



No. in plan	Title
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9	Indoor switchgear
10	Substation-shared control room
11.1, 11.2	Diesel-generator sets
12	Garage

Figure 5: General plan of the land-based NPP (Reproduced courtesy of OKBM Afrikantov)

SHELF (NIKIET, Russian Federation)

1. Introduction

A power unit based on the SHELF reactor is designed for use as a local power source for users in remote and hard-to-reach locations. The power unit, as shown in *Figure 1* is a power capsule to generate 6.4 MW(e). The power capsule accommodates all reactor components, the turbine generator package (TGP), the automated and remote control system, monitoring and protection system and the electricity output regulation. SHELF power capsule can be used for both floating and submerged nuclear power plants. Engineering design of SHELF is similar to a large degree with marine propulsion nuclear power plants. The power unit is delivered as a single item with all of its components accommodated inside a high-strength vessel (containment). This ensures high quality of the module fabrication at a specialized machine-building plant. The reactor is refueled, exhausted equipment is replaced, and the reactor, the TGP and other components are dismantled at the end of service life at a specialized plant. The capability for long-term unattended automated operation of the reactor, the TGP and other equipment inside the SHELF module eliminate the need of plant operators inside the power unit and to keep the module unmanned during the automated operation period equal to 5000 hours. After the automated operation, the scheduled maintenance is conducted within 15 days.

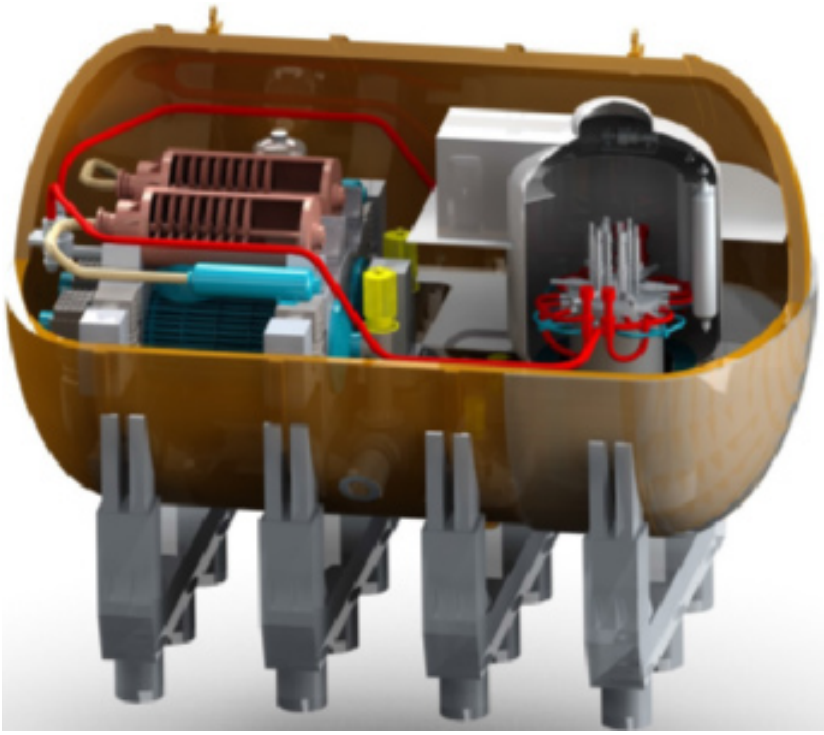


Figure 1: Power unit of an NPP with SHELF-10 reactor facility
(Reproduced courtesy of NIKIET)

2. Target Application

SHELF reactor is designed for use as a source of power for installations in remote and hard-to-reach locations with decentralized power supplies including the sites on Arctic shore regions. The electric power of the single-unit plant with SHELF reactor is 6.4 MW while

thermal power is 28 MW. Depending on the consumer requirements a SHELF plant can be equipped with a system for direct heat supply to resident and production premises with a capacity of 12 Gcal/h or a distillation desalination plant with a capacity of 500 m^3/h of fresh water. In case of land based plant, the heat is rejected by external heat exchangers with mechanical pumping of air. The SHELF plant does not require local water sources. Decay heat removal in case of submerged deployment is provided by sea water cooling.

3. Development Milestones

2012	A concept of an undersea power unit with the SHELF reactor developed by NIKIET. The request for proposal was issued
2015	Study of a land-based unit design and the development of a line of integral facilities in an electric power range from 1 to 8 MW.
2016	Detailed design of SHELF reactor unit with an electric power of 5.4 MW underway at NIKIET

4. General Design Description

SHELF is a water-cooled reactor of integral layout with a combined (forced and natural) coolant circulation modes. The fuel cycle is 6 years with the design capacity factor of 80%. The components for SHELF are accommodated inside a cylindrical power capsule with an inner diameter of 8 m and a length of 14 m. The reactor equipment is installed in the rear portion of the capsule and the TGP equipment is installed in its front. Besides, there is a compact module housing the auxiliary systems including an air conditioning system for module room temperature maintenance at a level of not more than 50°C. The unit's external systems include an automated instrumentation and control system (AICS), uninterrupted power supply systems, as well as the reactor facility and TGP auxiliary systems, including a ventilation system, a negative pressure system and others. Following the delivery to the operation site, the SHELF unit module is installed, secured on dedicated supports and connected to the other NPPs equipment through service lines. The modular approach allows the capital construction cost to be minimized and the requirements to the nuclear plant deployment site reduced.

Reactor Core

The core is of the heterogeneous cartridge type and consists of 163 hexahedral fuel assemblies (FA) of three different types. In a number of FAs burnable poison and control absorber rods are located. Absorber rods for reactivity control are united in six identical shim groups. The choice of fuel composition was governed by the maximum fuel load needed to provide the required reactor core life with the fuel enriched to not more than 20%; fuel composition consists of uranium dioxide in a silumin matrix in the form of cylindrical fuel elements. The specific advantages offered by this fuel type in comparison with the ceramic fuel of the container type applied in large NPP are as follows:

- A potentially higher nuclear fuel burn-up (up to 160 MW·day/kg);
- Low temperature in the fuel composition and, as a result, a small amount of accumulated heat;
- Serviceability ensured at different power levels in a combination with high manoeuvrability;
- An extra safety barrier due to the fuel matrix properties, ensured relative to the fission product release.

The reactor core size, composition and load are chosen so to provide the preset core life and capability conditions. Highly pure water is used as the core coolant and moderator. The ammonia addition is used to generate hydrogen in water with the aim to prevent corrosive

oxidative radiolysis products generation. The operating time of the reactor with one fuel load is 5.6 years, with no scheduled outages for maintenance included, and the reactor core life is 40000 hours. The reactor core exhibits negative temperature, density and power reactivity coefficients across the range of the reactor parameters variation. This factor is favourable for the core self-regulation and the safety improvement. Reactor refueling according to the adopted concept is performed on the specialized enterprise basis. In-service fuel handling is not envisaged.

Reactor coolant system

SHELF's thermal-hydraulic circuit consists of two self-sustained systems: nuclear steam supply system (NSSS) and a turbine generator system. NSSS comprises the following major systems:

- Primary circuit system;
- Secondary circuit system;
- Emergency core cooling system (ECCS);
- Emergency cooldown system (ECS);
- Makeup, dual and emergency absorber injection system;
- Equipment cooling system;
- Reactor overpressure protection system;
- SG overpressure protection system;
- Safeguard vessel and containment overpressure protection system;
- Instrumentation and control system (I&C).

SHELF is an integral-type reactor. This ensures the compactness of the system and equipment and improves the reliability of the facility due to a smaller number of supply lines subjected to the primary coolant pressure. The simplicity of the coolant line layout, combined with the integral reactor design, provides, when the coolant circulates naturally, for a high level of removed power being enough to keep the reactor operating at a power of up to 65% of the rated value in the event the pump power supply loss and for the accident-free core cooldown.

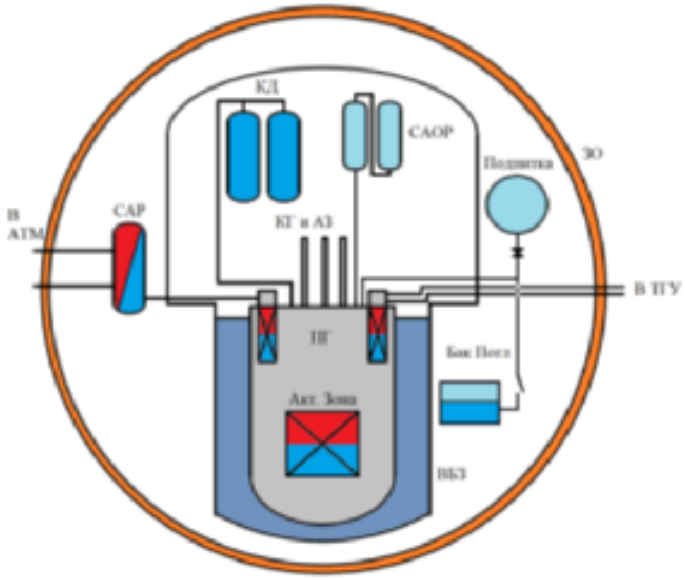


Figure 2: Key systems of the SHELF reactor facility (Reproduced courtesy of NIKIET)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	NIKIET, Research and Development Institute of Power
Country of origin	Russian Federation
Reactor type	Integral PWR
Electrical capacity (MW(e))	6.4
Thermal capacity (MW(th))	28
Expected capacity factor (%)	80
Continuous operation period (hr)	40000
Design life (years)	
Plant footprint (m^2)	N/A
Coolant/moderator	Light water
Primary circulation	Forced and natural circulation
System pressure(MPa)	–
Core inlet/exit temperatures (^oC)	–
Core flow rate (ton/hr)	–
Steam generating capacity (ton/hr)	–
Steam temperature (^oC)	–
Pressure (MPa abs)	–
Neutron fluence at the end of life time (neutron/cm^2)	–
Main reactivity control mechanism	Control Rod Drive Mechanism (CRDM)
RPV height (m)	14
RPV diameter (m)	8
Module weight (metric ton)	–
Configuration of reactor coolant system	Integral PWR
Power conversion process	–
Fuel type/assembly array	–
Fuel assembly active length (m)	–
Number of fuel assemblies	163
Fuel enrichment (%)	< 20
Fuel burnup (GWd/ton)	–
Fuel cycle (years)	5.6
Cogeneration capability	Yes
Approach to engineered safety systems	Combined active and passive
Number of safety trains	–
Refuelling outage (days)	–
Distinguishing features	Power source for users in remote and hard-to-reach locations; can be used for both floating and submerged nuclear power plants
Modules per plant	–
Target construction duration (months)	–
Seismic design (g)	–
Core damage frequency (per reactor-year)	–
Design Status	Detailed design underway

Pressurizer
The design calculations have shown that the primary circuit pressure does not exceed the specified level in the design-basis situations. The design includes an overpressure protection system as an additional engineering approach for the management of beyond design-basis accidents and serves to prevent, in such cases, the primary circuit system boundaries from breaking down or being loaded with forces in excess of those permitted. The system incorporates two parallel initiation lines. To confine radioactive products, the fluid is discharged over the reactor overpressure protection lines, via a bubbler, into the biological shielding inner tank inside the safeguard vessel. Apart from the primary circuit pressurizing system, the SHELF facility with an integral reactor contains independent pressurization and secondary circuit equipment overpressure protection systems, and the steam generator, safeguard vessel and containment overpressure protection systems.

Power conversion system
The SHELF two-circuit integral water-cooled reactor facility is an integration of systems and components intended to convert nuclear energy to thermal power. It includes a reactor and associated systems required for its normal operation, emergency cooling, emergency protection and maintenance in a safe condition. Auxiliary and support functions are performed by other SMR plants. The purpose of the primary circuit system is to remove heat from the reactor core and transfer it to the secondary circuit fluid in the SG. It comprises a main circulation line and a coolant inventory variation compensation system. The purpose of the secondary circuit system is to generate superheated steam from feedwater and transfer heat to the NPP's turbine generator plant. The reactor facility's secondary circuit system comprises a steam generator installed inside the reactor vessel and pipelines with valves (outside the vessel).

Control and Protection System
A monitoring, control and protection system (CPS) is used to perform the functions of the reactor facility's control and protection in all normal and emergency operation modes. The CPS is a part of the SHELF unit's automated instrumentation and control system (AICS). The CPS is designed to:

- Control the reactor core reactivity and power through the generation and performance of control actions on the reactor core in all operating modes of the reactor facility;
- To monitor the neutron flux (power), its variation rate, and the process parameters required for the protection and control of the reactor core reactivity and the reactor facility power;
- To function as the initiating portion (at the upper coordinating level) of the control safety systems and to regulate the reactor facility's key process parameters;
- To render and keep the reactor subcritical.
- The requirements to the CPS are defined both by Russian and international regulatory documents.

Steam Generator
The once-through steam generator (SG) is part of the SHELF reactor and is designed to generate superheated steam in the process of the reactor operation and to remove heat from the primary circuit system during the reactor cooldown. The SG comprises a tubing, collection and distribution chambers installed in the annulus above and below the tubes, steam and feedwater lines installed inside the reactor vessel, steam and water risers installed on the reactor vessel flange, and outside steam and feedwater lines with shutoff and isolation

valves. The SG's heat transfer surface is divided into four independent sections cut off, when required, by the shutoff and isolation valves.

5. Safety Features

High level of the SHELF reactor safety is achieved through the following aspects.

1. Use of an integral water-cooled water-moderated reactor with well-developed intrinsic self-protection properties and the following inherent features:
    - negative core reactivity coefficients across the operating range of parameters;
    - high level of natural coolant circulation, which ensures efficient cooling of and heat transfer from the core in beyond design-basis and design-basis accidents;
    - high heat accumulating capacity of metal structures and large coolant inventory in the reactor, which ensure relatively slow transients during accidents caused by the loss of normal heat removal from the core.
2. A defense-in-depth system of barriers to the spreading of ionizing radiation and radioactive products of uranium fission into the environment, as well as a combination of implemented engineering and organizational measures to protect these barriers from internal and external impacts. Safety barrier system includes:
    - Fuel matrix;
    - Fuel cladding;
    - Leak tight primary circuit – reactor vessel;
    - Safeguard vessel;
    - Confinement valves;
    - Containment.
3. Application of passive safety systems and features that operate based on the natural processes without energy supply from outside. Such systems are as follows:
    - Structure of CPS drive actuators;
    - Decay heat removal system (DHRS);
    - Emergency core cooling system (ECCS);
    - Safeguard vessel enables the reactor core to maintain primary coolant level and to remove heat under all serious accidents as well as the radioactive product confinement when the primary circuit leak tightness is lost;
    - Containment that restricts the radioactive product release when the safeguard vessel is opened for some activities and in case of beyond design-basis accidents;
    - Reactor, safeguard vessel and containment protection systems equipped with membrane and rupture devices that operate under direct fluid action when a pressure exceeds the allowed limit.
4. Safety system reliability:

High reliability level of safety systems is achieved through implementing the following principles:

- Passive operation not requiring any actions for initiation;
- A diversity of safety systems and devices achieved through the use of different operating principles of systems (e.g., use of the CPS electromechanical drives for the emergency shutdown of the reactor, these being functionally divided into reactivity compensation member drives and emergency protection drives, as well as of the liquid absorber injection system);

- Redundancy of safety systems;
- Continuous and periodic monitoring of the in-service status of systems and components.

5. Protection against external impacts:

The reactor facility's containment ensures that the reactor components inside the containment and the safeguarded vessel are not damaged in the event of external impacts, including typhoon, hurricane, snow and icing, as well as of helicopter or airplane impacts on the SHELF NPP.

6. Plant Operational Performance

The electric power of one SHELF unit is 6.4 MW(e), and the thermal power is 28 MW(th). The current supplied to the consumer system is alternate and three-phase (voltage 0.4 kV ± 2%, frequency 50 Hz ± 1 Hz). The nuclear plant base operation mode is power operation in a range from 20 to 100% full power with the capability to vary the consumed power daily and annually. The power increase and decrease rate is 1% (forced primary coolant circulation) and 0.3% (natural primary coolant circulation). The time of the reactor operation with one fuel load is 40000 effective hours.

7. Instrumentation and Control System

The automated process control system (APCS) of a nuclear plant with the SHELF-10 reactor is intended to control the major and auxiliary electricity generation processes in all modes of the unit operation:

1. Normal operation:
   - Phased automated initiation;
   - Operation at steady power levels in a range of 20 to 100% Nnom with forced primary coolant circulation;
   - Operation at steady power levels in a range of 20 to 95% Nnom with natural primary coolant circulation;
   - Switchovers from one steady power level to another in the above power ranges at a preset rate (manoeuvring);
   - Switchover from natural primary coolant circulation to forced circulation;
   - Scheduled automated deactivation.
2. Anticipated operational occurrences:
   - Emergency power reduction and operation with a decreased steam supply due to failures of the reactor facility's key components or feedwater supply and steam receipt systems;
   - Deviation of the reactor facility's parameters to beyond the preset limits, leading to the need for the operator intervention or automatic controlling actions to normalize the parameters.
3. Emergency: Emergency deactivation in the event of reactor facility parameters deviating beyond the safe operation limits or in the event of equipment failures leading to the safe operation limits being violated.

8. Plant Arrangement

The undersea power unit module is an energy capsule which accommodates all components of the reactor facility, the TGP, and the unit equipment automated and remote control,

monitoring and protection systems, including the electricity output regulation, monitoring and control equipment. The unit's land based installation includes the AICS equipment, the AICS undersea equipment uninterruptible power supply systems, as well as auxiliary systems to support the reactor facility and TGP operation, including the ventilation system, the negative pressure system and others. The power supply system arrangement is selected based on the deployment conditions and the consumed auxiliary power with regard for the supporting utilities and physical protection systems. The power generation system consists of a generator connected to a steam turbine, generator current distributors and switchgear. The plant, including its power generation and supply system, and the auxiliary power system, is operated automatically with control from the APCS.

9. Design and Licensing Status

The licensing of the nuclear plant design is scheduled for 2017-2019.

10. Plant Economics

The current estimated performance indicators prove the nuclear plant with the SHELF reactor facility to be highly competitive not only in comparison with similar nuclear plant designs but also as compared to conventional generation based on imported fuel. The estimated payback period for a single-unit nuclear plant with the SHELF reactor facility is 14 years.

HIGH TEMPERATURE GAS COOLED SMALL MODULAR REACTORS

HTR-PM (Tsinghua University, China)

1. Introduction

In 1992, the China Central Government approved the construction of the 10 MW pebble bed high temperature gas cooled test reactor (HTR-10) in Tsinghua University's Institute of Nuclear and New Energy Technology (INET). In 2003, the HTR-10 reached its full power operation. After that, INET has completed a lot of experiments on the HTR-10 to verify crucial inherent safety features of modular HTRs, including:

- Loss of off-site power without scram;
- Main helium blower shutdown without scram;
- Withdrawal of control rod without scram;
- Helium blower trip without closing outlet cut-off valve.

The second step of HTR development in China began in 2001 when the high-temperature gas-cooled reactor pebble-bed module (HTR-PM) project was launched. The first concrete of the HTR-PM demonstration power plant was poured on December 9, 2012, in Rongcheng, Shandong Province, as shown in *Figure 1(a).* According to its planned schedule, the power plant will start power generation in 2017 [1]. The civil work of the nuclear island's buildings has been finished as shown in *Figure 1(b),* and the first of two reactor pressure vessels was installed in March 2016.

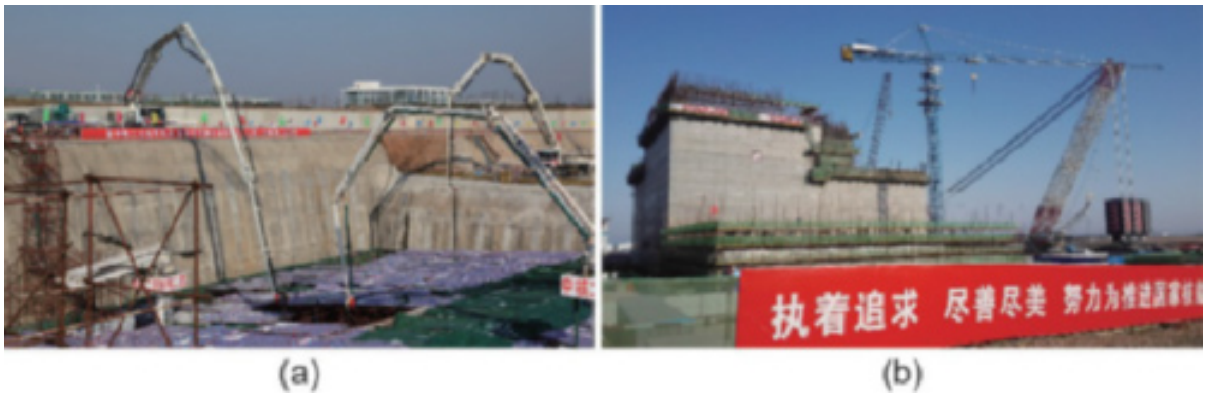


Figure 1: Construction of HTR-PM on (a) December 9, 2012 and (b) May 25, 2015 (Reproduced courtesy of INET)

2. Target Applications

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

Following HTR-PM demonstration plant, commercial deployment of HTR-PM based on batch construction is foreseeing, and units with more modules and bigger power size are under design. Units with multiple standardized reactor modules coupling to one single steam turbine (such as 200, 600 or 1000MW) are envisaged.

A conceptual design has been finished for a 600 MW(e) multi-module HTR-PM nuclear power plant, which consists of six reactor modules coupling to one steam turbine. Each

reactor module has the same design as the HTR-PM demonstration plant, with independent safety systems and shared non-safety auxiliary systems. The footprint of a multi-module HTR-PM plant is not remarkably different from that of a PWR plant generating the same power. *Figure 2* shows a 2 × 600 MW(e) HTR-PM nuclear power plant for cogeneration. Research and development on process heat applications, hydrogen production and gas turbines are continuing for future application

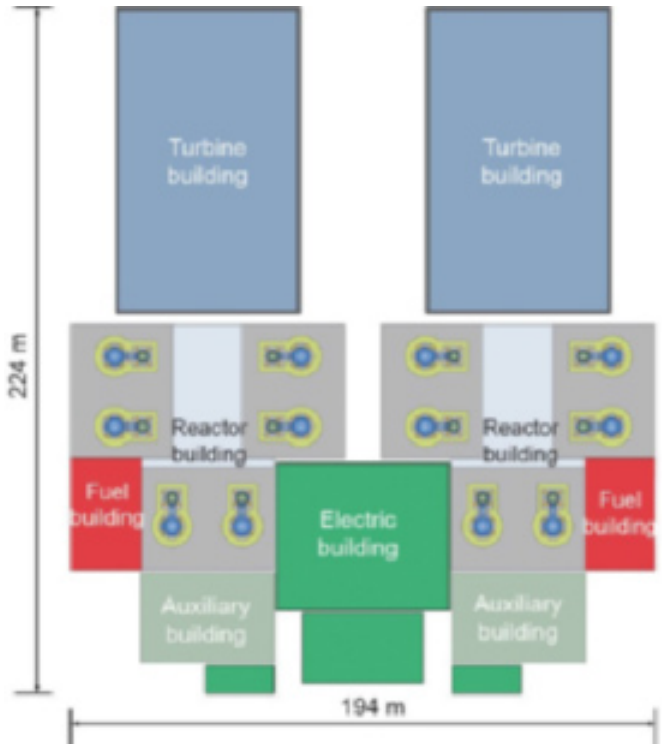


Figure 2: The 2 × 600 MW(e) HTR-PM multi-modules plant (Reproduced courtesy of INET)

3. Development Milestones

2001	Launch of commercial HTR-PM project
2004	Standard design of HTR-PM started
2006	HTR-PM demonstration power plant approved as one of National Science and Technology Major Projects
2006	Huaneng Shandong Shidaowan Nuclear Power Co., Ltd, the owner of the HTR-PM, established by the China Huaneng Group, the China Nuclear Engineering group Co. and Tsinghua University
2006-2008	Basic design of HTR-PM completed
2009	Assessment of HTR-PM PSAR completed
2012	First Pour of Concrete of HTR-PM
2013	Fuel plant construction started
2014	Qualification irradiation tests of fuel elements completed
2015	Civil work of reactor building finished
2016	RPV and core barrel etc. delivered, installation of main components ongoing
2017	First operation expected

4. General Design Description

The HTR-PM should achieve the following technical goals: (1) demonstration of inherent safety features; (2) demonstration of economic competitiveness; (3) confirmation of proven technologies; (4) standardization and modularization.

Design Philosophy

The HTR-PM consists of two pebble-bed reactor modules coupled with a 210 MW(e) steam turbine [2], as shown in *Figure 3*. Each reactor module includes a reactor pressure vessel (RPV); graphite, carbon, and metallic reactor internals; a steam generator; and a main helium blower. The thermal power of each reactor module is 250 MW(th), the helium temperatures at the reactor core inlet/ outlet are 250/750 °C, and steam at 13.25 MPa/567 °C is produced at the steam generator outlet.

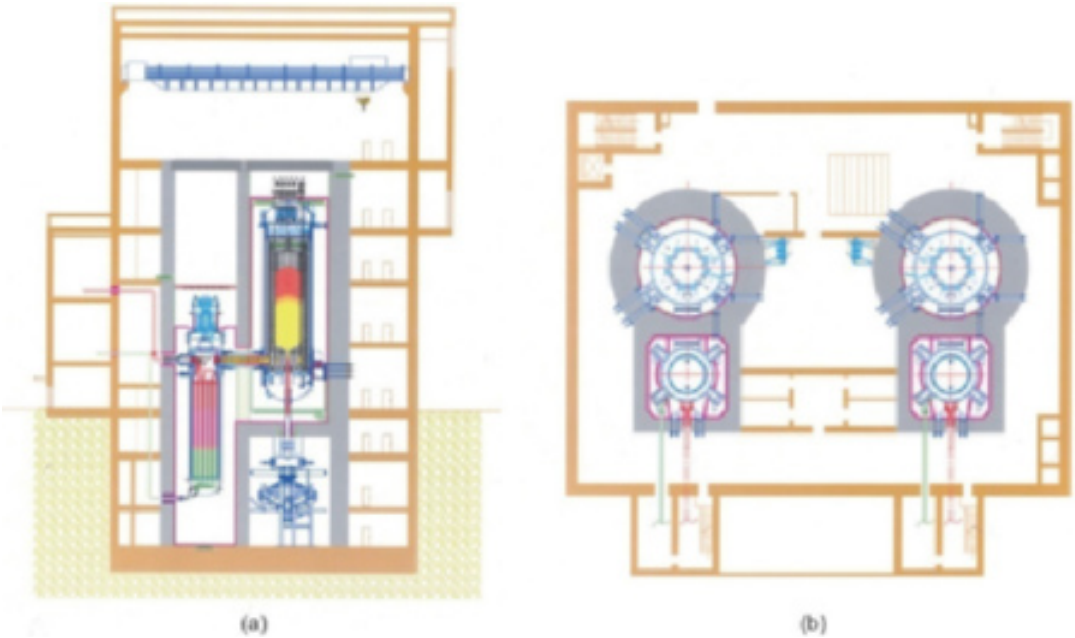


Figure 3: The HTR-PM demonstration nuclear power plant: (a) Front view; (b) top view (Reproduced courtesy of INET)

Reactor Core and Power Conversion Unit

The primary helium coolant works at the pressure of 7.0MPa. The rated mass flow rate is 96kg/s. Helium coolant enters the reactor in the bottom area inside the pressure vessel with an inlet temperature of 250ºC. Helium coolant flows upward in the side reflector channels to the top reflector level where it reverses the flow direction and flow into the pebble bed in a downward flow pattern. Bypass flows are introduced into the fuel discharge tubes to cool the fuel elements there and into the control rod channels for control rods cooling. Helium is heated up in the active reactor core and then is mixed to the average outlet temperature of 750ºC and then flows to the steam generator.

As shown in *Figure 4*, the reactor core and the steam generator are housed in two steel pressure vessels which are connected by a connecting vessel. Inside the connecting vessel, the hot gas duct is designed. All the pressure retaining components, which comprise the primary pressure boundary, are in touch with the cold helium of the reactor inlet temperature.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology Developer:	INET, 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):	~60000 (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
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, inherent safety by large negative temperature coefficients, large heat capacity
Active Safety Features:	Yes, control rod insertion with SCRAM; Turbine trip
Fuel Type/Assembly Array:	Spherical elements with coated particle fuel
Fuel Pebble Diameter (cm):	6
Number of Fuel Spheres:	420000 (in each reactor module)
Fuel Enrichment (%):	8.5
Fuel Burnup (GWd/ton):	90 (in average while discharge)
Fuel Cycle (months):	N/A; Online / on-power refuelling Fuel remain in the reactor for ~35 months
Emergency Safety Systems:	Control rod insertion; Circulator trip; isolation of secondary circuit; drain of steam generator in the case of steam generator
Residual Heat Removal Systems:	Passive
Refuelling Outage (days):	N/A since online
Distinguishing Features:	Inherent safety, no need for offsite emergency measures
Modules per Plant:	Two modules with their own steam-generator feeding one turbine-generator set.
Estimated Construction Schedule (months):	59 months for the demonstration power plant
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.

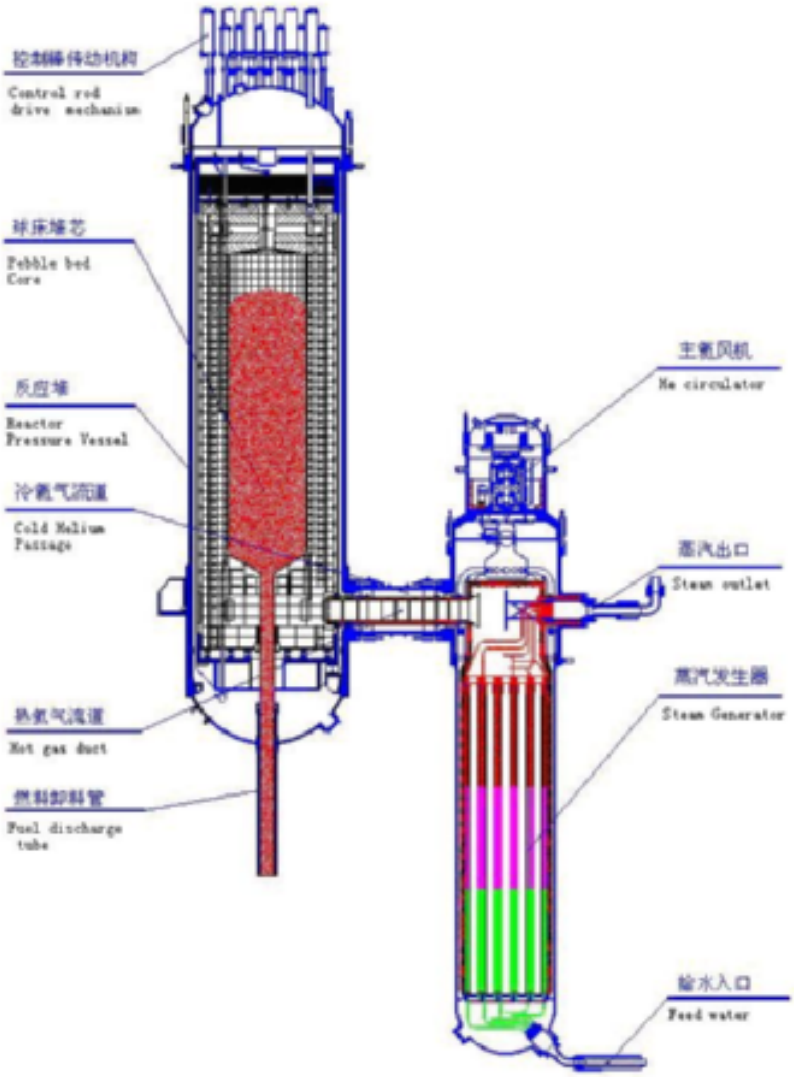


Figure 4: HTR-PM primary loop (Reproduced courtesy of INET)

Fuel characteristics

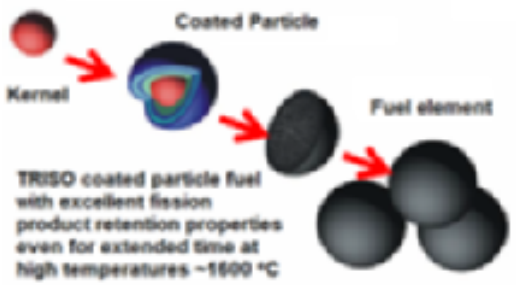


Figure 5: Spherical Fuel Elements (Reproduced courtesy of INET)

Fuel elements are spherical ones. Every fuel element contains 7g heavy metal. The enrichment of U-235 is 8.5%. Uranium kernels of about 0.5mm diameter are coated by three layers of pyro-carbon and one layer of silicon carbon. Coated fuel particles are dispersed in matrix graphite which is 5cm in diameter. Surrounding the fuel containing graphite matrix is a 5mm thick graphite layer

Fuel handling system

The operation mode of HTR-PM adopts continuous fuel loading and discharging: the fuel elements drop into the reactor core from the central fuel loading tube and are discharged through a fuel extraction pipe at the core bottom. Subsequently, the discharged fuel elements pass the burn-up measurement facility one by one. Depending on their state of burn-up they

will either be discharged and transported into the spent fuel storage tank while running up to the design threshold of burn-up, or be re-inserted into the reactor to pass the core once again.

Reactivity Control
Two independent shutdown systems are installed: a control rod system and a small absorber sphere (SAS) system, both placed in holes of the graphitic side reflector. Reactivity control is achieved through 24 control rod assemblies, and six SAS shutdown systems serve as a reserve shutdown system. The control rods are used as a regulating group that is used during normal plant operation to control reactivity, but also for hot shutdown. The SAS system is used for long-term shutdown. Absorber material of control rods and small absorbers is B_4C.

Reactor Pressure Vessel and Internals
The primary pressure boundary consists of the reactor pressure vessel (RPV), the steam generator pressure vessel (SGPV) and the hot gas duct pressure vessel (HDPV), which is all housed in a concrete shielding cavity. The three primary pressure vessels are composed of SA533-B steel as the plate material and (or) the 508-3 steel as the forging material. The ceramic structures surrounding the reactor core consist of the inner graphite reflector and outer carbon brick layers. The whole ceramic internals are installed inside a metallic core barrel, which itself is supported by the RPV. The metallic core barrel and the pressure vessel are protected against high temperatures from the core by the cold helium borings of the side reflector, which act like a shielding temperature screen.

5. Plant Safety Features

Description of safety concept
The HTR-PM will realize the following safety features:
(1) The radioactive inventory in the primary helium coolant is very small when the reactors are working at normal operation conditions. Even if this limited amount of radioactivity would be released into the environment following an accident, there is no need to take any emergency measures. (2) For any reactivity accident or for a loss of coolant accident, the rise of the fuel element temperature will not cause a significant additional release of radioactive substances from the fuel elements. (3) The consequences of water or air ingress accidents depend on the quantity of such ingresses. The ingress processes and the associated chemical reactions are slow, and can readily be terminated within several dozens of hours (or even some days) by taking very simple actions.

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 (due to 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 [3].

Another feature of the HTR-PM design is the long time period of accident progression due to the large heat capacity of fuel elements and graphite internal structures. It requires days for the fuel elements to reach the maximum temperature when the coolant is completely lost.

For the HTR-PM plant, the most significant design basis accident is the complete loss of primary helium coolant. In such an accident scenario, no any additional cooling systems are designed for cooling the reactor core. The decay heat in fuel elements shall dissipate to the outside of the pressure vessel primarily by means of heat conduction and radiation within the reactor core internals. The maximum fuel temperature during this accident scenario is the highest and most challenging for the fuel, therefore this accident scenario is the most critical one for the HTR-PM design. In the case of HTR-PM, the maximum fuel temperature will be less than 1620°C for any conceivable accident.

Engineered Safety System Configuration and Approach

When accidents occur, a very limited number of reactor protection actions shall be called upon by the reactor protection system. No or very limited actions through any systems or human interventions are foreseen after the limited reactor protection actions are activated. The limited reactor protection actions shall be to trip the reactor and the helium circulator, to isolate the primary and secondary systems. When there is large leak or rupture of steam generator heat exchange tubes, a water discharge system is designed to minimize the amount of water ingress into the reactor core.

Reactor Cooling Philosophy

Normally the reactor is cooled by steam generating system.

Under accident conditions, the primary helium circulator shall be stopped automatically. Because of the low power density and the large heat capacity of the graphite structures, the decay heat in the fuel elements can dissipate to the outside of the reactor pressure vessel by means of heat conduction and radiation within the core internal structures, without leading to unacceptable fuel temperature limits. The decay heat shall be removed to heat sink passively by reactor cavity cooling system (RCCS). Even if the RCCS fails, the decay heat can be removed transferring through the concrete structure of reactor cavity while the temperatures of fuel elements are under design limit.

Containment Function

Retention of radioactivity materials is achieved through multi-barriers. The fuel elements with coated particles serve as the first barrier. The fuel elements used for HTR-PM have been demonstrated to be capable of retaining fission products within the coated particles under temperatures of 1,620ºC which is not expected for any plausible accident scenarios. The second barrier is the primary pressure boundary which consists of the pressure vessels of the primary components. The vented low pressure containment (VLPC) is designed according to ALARA principle to mitigate the influence of accidents, consisting of the reactor building and some additional auxiliary buildings which house primary helium containing components.

6. Plant Operational Performances

The HTR-PM demonstration power plant is under construction. Due to online fuel loading mode of HTR-PM, better availability factor can be expected compared with other power plants operating in a mode of periodic fuel loading.

7. Instrumentation and Control Systems

The instrumentation and control system of HTR-PM is similar to those of normal PWR plant. The two reactor modules are controlled in a coordinated manner to meet various operational requirements.

8. Plant Arrangement

The nuclear island contains reactor building, nuclear auxiliary building, spent fuel storage building and I&C building, as shown in *Figure 6*. The steam turbo-generator, which is similar to that of a conventional fossil-fired power plant, is housed in the turbine building.

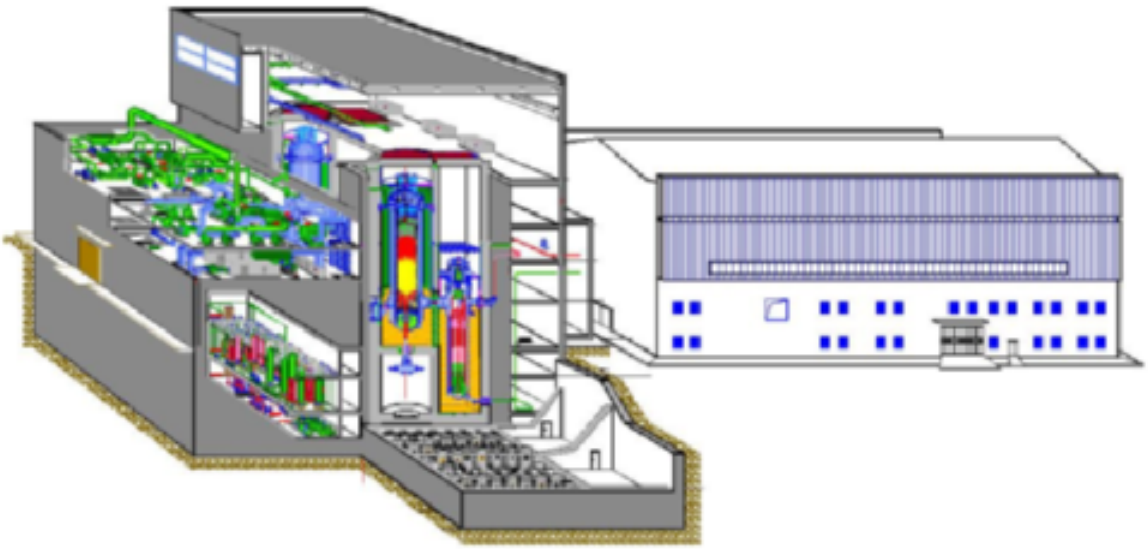


Figure 6: Nuclear island and conventional island layout (Reproduced courtesy of INET)

9. Design and Licensing Status

The preliminary safety analysis report (PSAR) was reviewed by the licensing authorities during 2008-2009. The Construction Permit was issued in December 2012.

10. Plant Economics

The HTR-PM demonstration power plant is a first-of-a-kind project in the world. Based on the cost data of the demonstration plant, a preliminary estimation shows that the cost of following six-module HTR-PM power plant is a little higher than that of a PWR plant generating the same power. However, the difference is only around 10~20% [4].

[1] Zhang, ZY; Dong, YJ; Li, F; Zhang, ZM; Wang, HT; Huang, XJ; Li, H; Liu, B; Wu, XX; Wang, H; Diao, XZ; Zhang, HQ; Wang, JH. The Shandong Shidao Bay 200 MW(e) High-Temperature Gas-Cooled Reactor Pebble-Bed Module (HTR-PM) Demonstration Power Plant: An Engineering and Technological Innovation. Engineering 2 (2016) 2(1): 112–118.

[2] Zhang, ZY; Wu, ZX; Wang, DZ; Xu, YH; Sun, YL; Li, F; Dong, YJ. Current status and technical description of Chinese 2 x 250 MW(th) HTR-PM demonstration plant. NUCLEAR ENGINEERING AND DESIGN, (2009) 239(7) : 1212-1219

[3] https://www.technologyreview.com/s/600757/china-could-have-a-meltdown-proof-nuclear-reactor-next-year/

[4] Zhang, ZY; Sun, YL. Economic potential of modular reactor nuclear power plants based on the Chinese HTR-PM project. NUCLEAR ENGINEERING AND DESIGN, (2007)237: 2265–2274

GTHTR300 (Japan Atomic Energy Agency, Japan)

1. Introduction

GTHTR300 (Gas turbine High Temperature Reactor 300 MW(e)) is a multi-purpose, inherently-safe and site-flexible SMR (small modular reactor) that Japan Atomic Energy Agency (JAEA) is developing for commercialization in around 2020s [1]. 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°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 a 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 [2]. 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°C, creating a large temperature difference from ambient air making the 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.

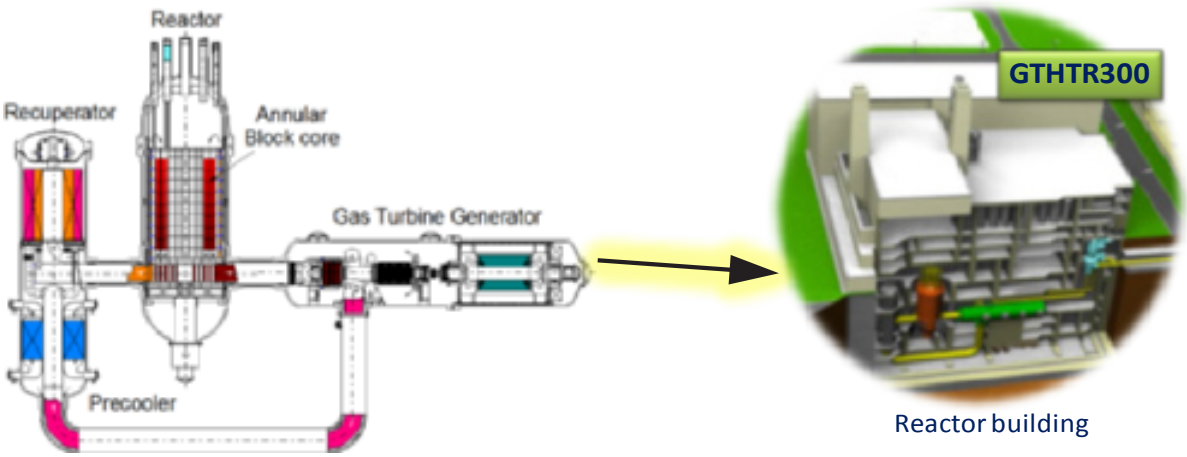


Figure 1: Reactor System Configuration of GTHTR300 (Reproduced courtesy of JAEA)

2. Target Applications

Typical applications include electric power generation, thermochemical hydrogen production [3], desalination cogeneration using waste heat only, and steelmaking. The reactor thermal power may be rated up to 600 MW(th) 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 55000 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.

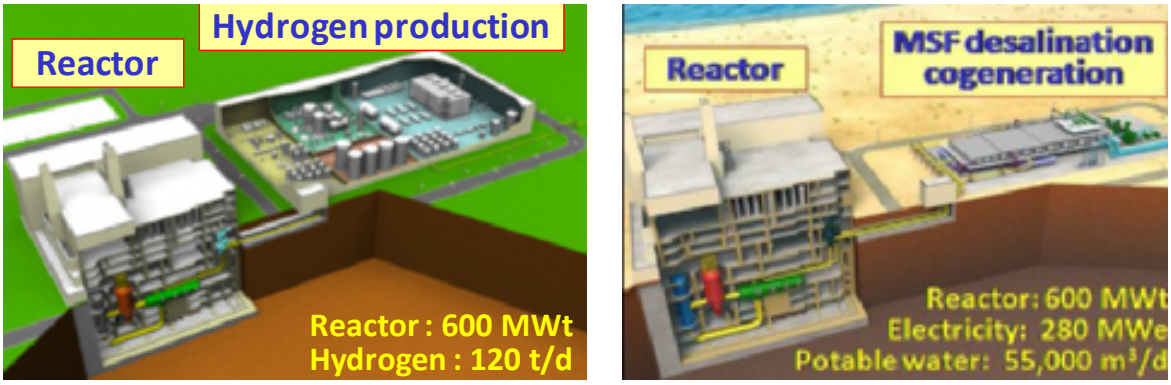


Figure 2: Cogeneration plant layout for hydrogen (left) and for desalination (right) (Reproduced courtesy of JAEA)

3. Development Milestones

2003	Basic design
2004	Design development start
2005	Cogeneration design (GTHTR300C)
2014	IS process continuous H2 production test facility construction
2015	Basic design for HTTR-connected gas turbine and hydrogen test plant (HTTR-GT/H2)
2020-25	HTTR-GT/H2 test operation (planned)
2025	Construction of lead commercial plant (planned)

4. General Design Description

Design Philosophy

The overall goal of the commercial plant design is to provide a family of system options capable of producing competitive electricity, hydrogen, desalination, other products, and yet deployable in the near term. The development of the multiple systems simultaneously does not necessarily suggest having investment and risk multiplied. Rather, the development requirement is minimized thanks to a design philosophy of system simplicity, economic competitiveness and originality, namely the SECO philosophy. There are three major elements to this design philosophy.

The first element is technology simplification. All design variants are built on the premise that they share common system technologies to the maximum extent possible. As a result, the design variants share a unified reactor and primary coolant circuit, an aerodynamically and mechanically similar line of helium gas turbines used for electricity production, and the IS process selected to produce hydrogen [4]. The helium gas turbine and the IS process are compatible application systems with the high temperature reactor heat source to enable economically competitive energy production.

The second element of the SECO design philosophy has been incorporating unique design attributes that are less demanding on the system technologies required. The efforts in this area have resulted in such original design simplification as conventional steel reactor pressure vessel construction, horizontal gas turbine installation, and system modular arrangement among others as shown in *Figure 3*. The technical design features of the GTHTR300 plant variants in greater detail.

The third element that has been made possible by constant pursuit of technology and design simplification is a focused technological development scope that comes with low risk and investment of overall development. Furthermore, since the technologies to be developed are shared by several systems, the benefit of investing in any one development is increased. On the site where the HTTR is constructed for acquiring the reactor technology, JAEA has also been carrying out research and development on the helium gas turbine and the IS process.

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, housing the reactor core, the gas turbine, and the 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.

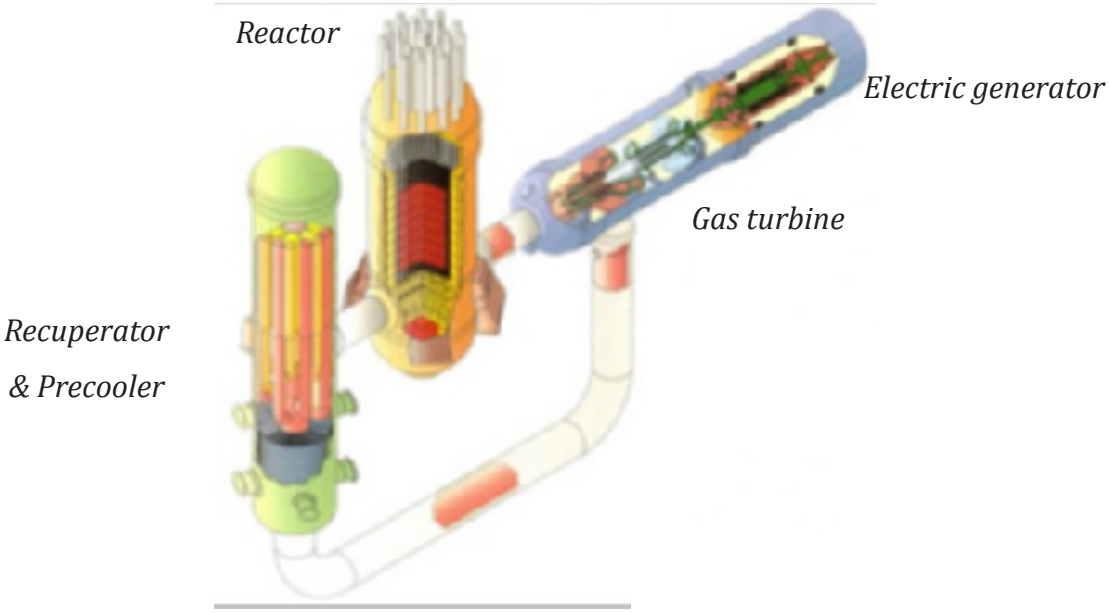


Figure 3: Functionally-oriented pressure vessel units of the reactor primary system (Reproduced courtesy of JAEA)

Fuel Design

The fuel design is coated fuel particle of less than 1 mm in diameter as shown in *Figure 4*. Each particle consists of a UO2 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.

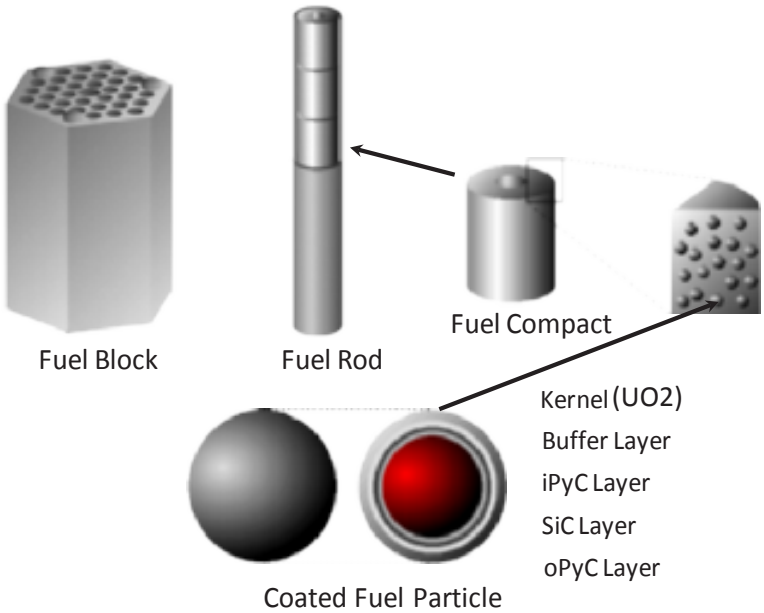


Figure 4: Fuel design features (Reproduced courtesy of JAEA)

Design Development

While the reactor technologies required for the GTHTR300 are developed mainly with construction and operation of JAEA's 30 MW(th) 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.

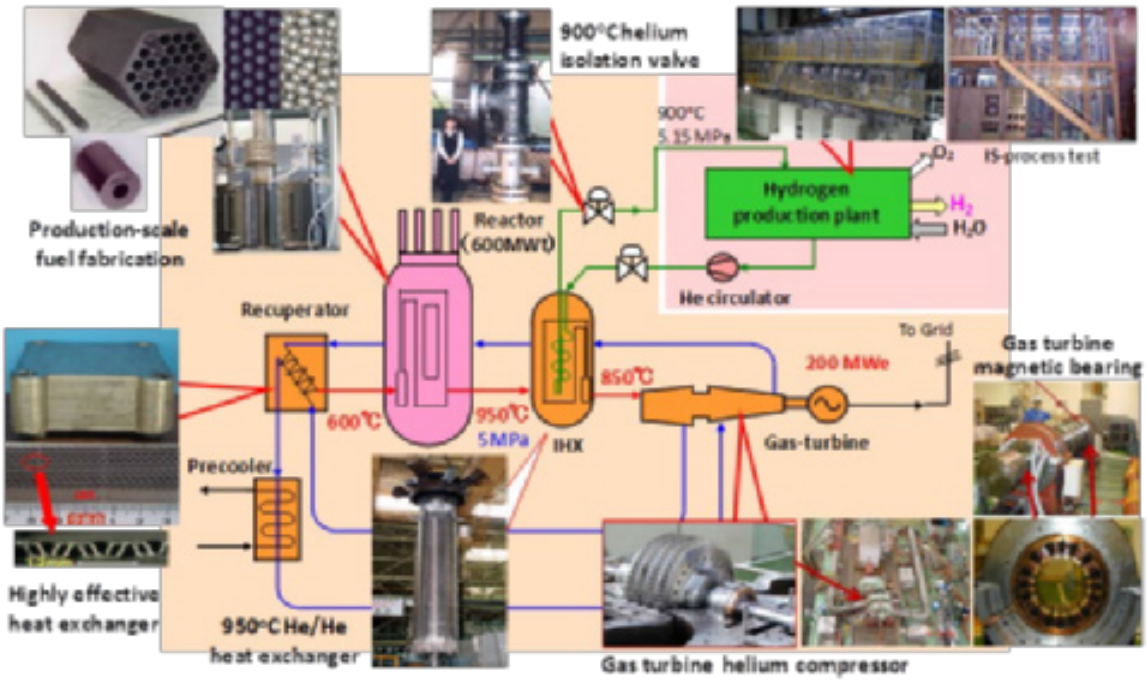


Figure 5: Plant technologies developed for GTHTR300 (Reproduced courtesy of JAEA)

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology developer	JAEA jointly with MHI, Toshiba/IHI, Fuji Electric, KHI, NFI, etc.
Country of origin	Japan
Reactor type	Prismatic HTGR
Electrical capacity (MW(e))	100~300 MW(e)
Thermal capacity (MW(th))	< 600 MW(th)
Design capacity factor	>90%
Design life (years)	60
Coolant	helium
Moderator	graphite
Primary circulation	forced circulation
System pressure	7 MPa
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
Refueling 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	>0.18 g automatic shutdown
Predicted core damage frequency per reactor year	<1 E-8/
Design Status	Basic design

5. Plant 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 decay heat of the reactor core can be removed 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, while the fuel temperature will remain below the fuel design limit as illustrated in *Figure 6*.

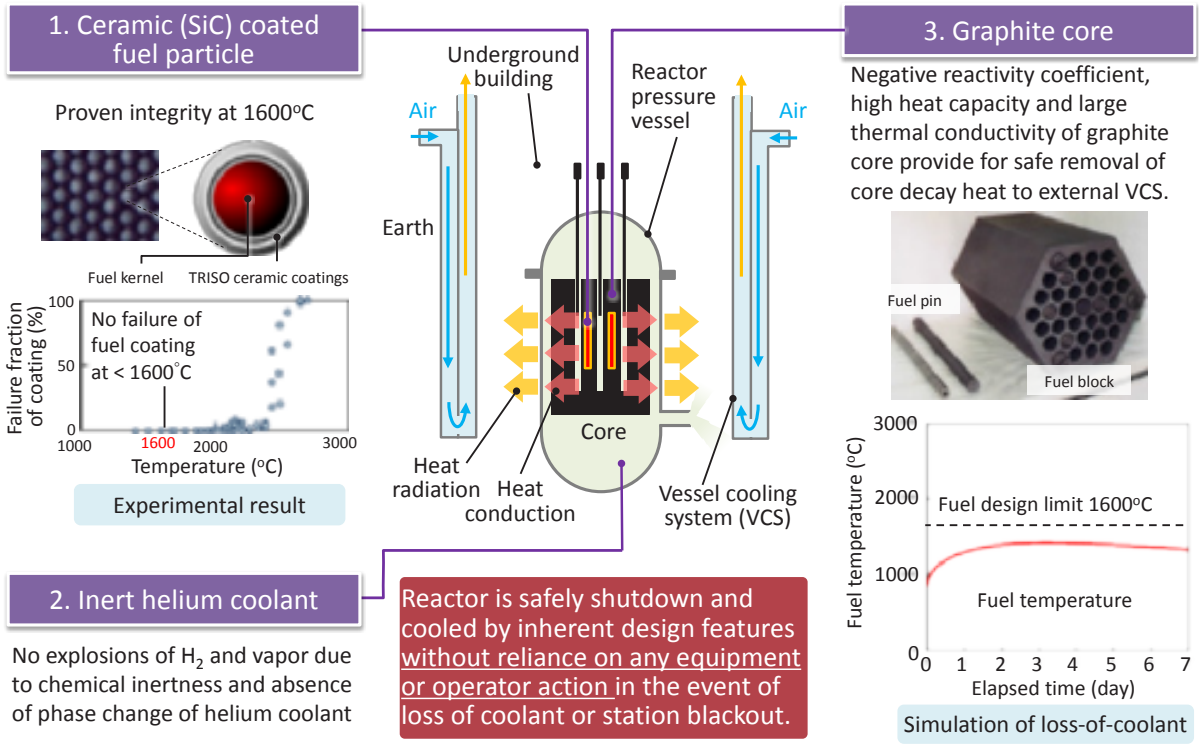


Figure 6: Inherent reactor safety design (Reproduced courtesy of JAEA)

6. Plant Operational Performances

The ability to follow variable power and heat loads [5] is simulated as shown in *Figure 7*. The simulation examines the plant response to an electric demand increase of 5% of the base rate per minute with corresponding reduction of the heat rate, which is the maximum requirement for cogeneration load follow. The reactor remains at 100% power at all times. Starting from the base cogeneration rates, the turbine power generation is increased to follow the electric load demand increase by increasing the primary coolant inventory through the inventory control valve IV1. The IHX heat rate to the thermal production plant is lowered by lowering the intermediate loop flow circulation rate with the variable speed gas circulator. As the primary exit temperature of the IHX begins to rise, the valve CV4 is opened, by active or prescheduled control to follow load demand, to direct cold flow from the compressor

discharge to mix with the hot exit gas of the IHX primary side. By applying flow bypass via CV4 the goal to maintain the turbine inlet temperature near the rated 850°C is achieved, as shown in *Figure 8*. The power sent out to external grid increases to 276 MW(e) from 178 MW(e) in as little as 7 minutes. The pressure in the reactor and at turbine inlet increases to 7 MPa from 5 MPa. To return to the base cogeneration state, the control is reversed by reducing primary coolant inventory through another inventory control valve IV2 and simultaneously by closing the bypass valve VC4.

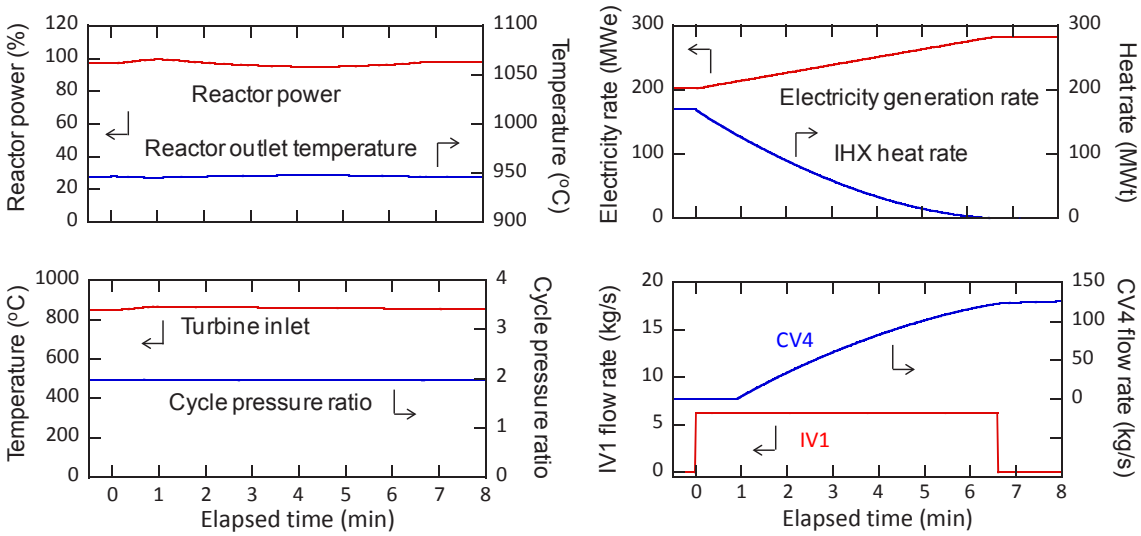


Figure 7: Simulation of cogeneration plant load follow to +5%/min electric load increase (Reproduced courtesy of JAEA)

7. Instrumentation and Control System

Figure 8 illustrates the plant control system and instrumentation needed to enable load following in power generation mode as well as in cogeneration mode. The overall approach to dynamic operation integrates the following four load control strategies:

- Control of turbine speed, Sd, through flow bypass valve CV1.
- Control of recuperator low-pressure-side inlet temperature, Tx, through flow bypass valve CV2.
- Control of turbine inlet temperature, Tt, by flow bypass valve CV3.
- Control of turbine inlet temperature and pressure, Tt and Pt, by bypass valve CV4, and inventory flow valves IV1 and IV2.

The first two strategies are used to control rapid transients such as a sudden loss of electric generator load. They are effective to protect the gas turbine from excess overspeed and prevent thermal shock in the recuperator.

The third strategy is used to automate heat rate to follow slow or fast changes in the heat load of the IHX when it is perturbed by the thermal production plant. As the IHX primary exit flow temperature rises or falls in response to a change in the IHX secondary heat load, the flow valve CV3 opens or closes to introduce more or less cold flow upstream of the turbine (from the compressor discharge to the turbine inlet) to keep the turbine inlet temperature constant. The overall control strategy aims to continue normal power generation, unaffected by any heat load change in the IHX.

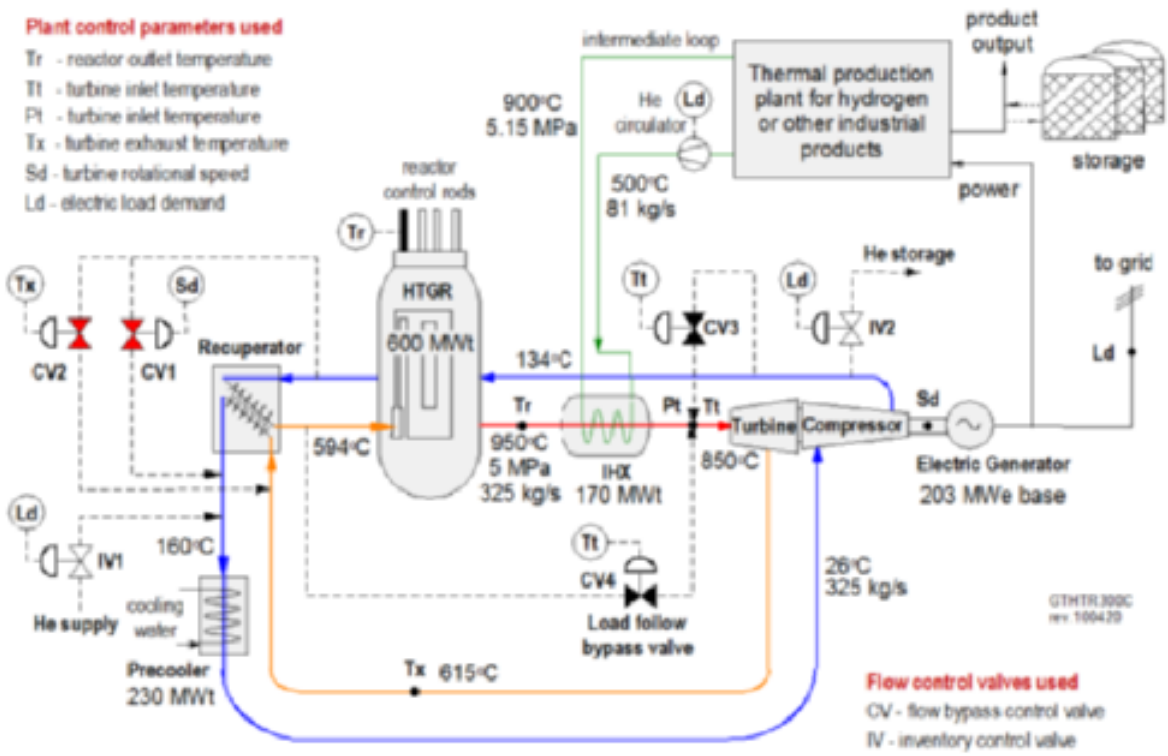


Figure 8: Control system for GTHTR300 (Reproduced courtesy of JAEA)

The fourth control strategy is applied to automate cogeneration load follow. The conditions to be met include (1) constant reactor temperature to avoid thermal stress in high temperature structure; (2) constant reactor thermal power to yield base load economics; and (3) constant power generation efficiency over a broad range of load follow.

8. Plant Design and Licensing 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.

[1] X. Yan, K. Kunitomi, T. Nakada And S. Shiozawa, "GTHTR300 design and development", Nucl. Eng. Des., 222, 247 (2003).

[2] H. Sato, X. Yan, Y. Tachibana, K. Kunitomi, "GTHTR300—A nuclear power plant design with 50% generating efficiency", Nucl. Eng. Des., 275, 190 (2014)

[3] K. Kunitomi, X. Yan, T. Nishihara, N. Sakaba And T. Mouri, "JAEA's VHTR for hydrogen and electricity cogeneration: GTHTR300C", Nuclear Engineering and Technology, 39, 1, 9 (2007).

[4] S. Kasahara, N. Tanaka, H. Noguchi, J. Iwatsuki, H. Takegami, X. L. Yan And S. Kubo, "JAEA's R&D on the thermochemical hydrogen production IS process", HTR2014, Weihai, China, 27-31 Oct., Paper HTR2014-21233, Institute of Nuclear and New Energy Technology (2014).

[5] X. Yan, H. Sato, Y. Tachibana, K. Kunitomi, R. Hino, "Evaluation of High Temperature Gas Reactor for Demanding Cogeneration Load Follow", J. Nucl. Sci. Technol., 49, 1, 121 (2012)

GT-MHR (OKBM Afrikantov, Russian Federation)

1. 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 manoeuvring characteristics of the reactor plant (RP) for regional power sources.

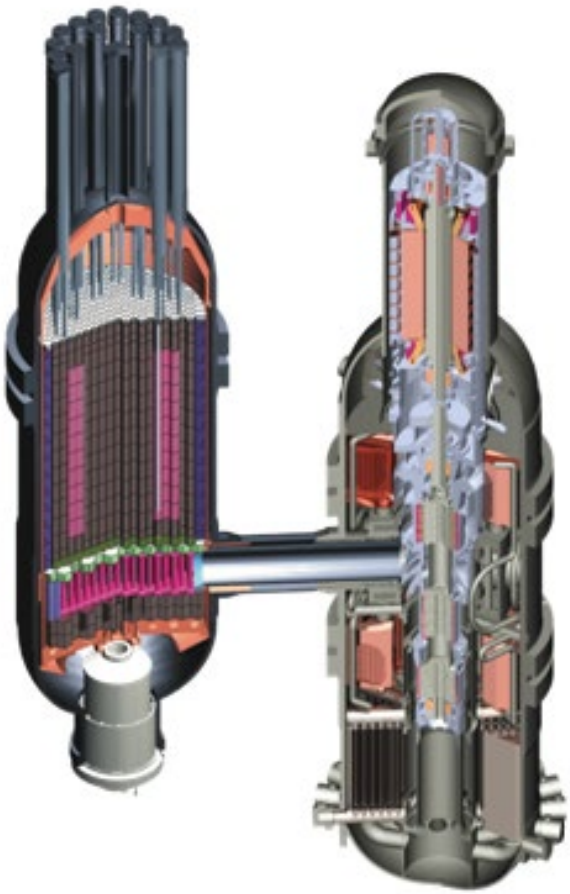


Figure 1: Reactor System Configuration of GT-MHR
(Reproduced courtesy of OKBM Afrikantov)

2. Target Application

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.

3. 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

4. General Design Description

Design Philosophy

The GT-MHR direct Brayton cycle power conversion system contains a gas turbine, an electric generator and gas compressors. The layout can be seen in *Figure 1*. 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.

Reactor Core and Fuel Characteristics

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°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.

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)	9110
Coolant/moderator	Helium
Primary circulation	Forced circulation
System pressure (MPa)	7.24
Core inlet/exit temperatures (^oC)	490 / 850
Main reactivity control mechanism	Rod insertion
RPV height (m)	31
RPV diameter (m)	10.2
RPV or module weight (metric ton)	N/A
Configuration of reactor coolant system	Loop/Integral type
Power conversion process	Direct Brayton or Indirect Rankine cycle
Fuel type/assembly array	Hexagonal Prism graphite block of 0.36m coated particle fuel
Fuel assembly 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
Cogeneration capability	Yes
Approach to engineered safety systems	Hybrid (active + passive)
Number of safety trains	2
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)
Target construction duration (months)	48
Seismic design	8 points (MSK 64)
Core damage frequency (per reactor-year)	Core damage frequency not applicable to HTGRs BDBA frequency < 1E-5 /year Frequency of ultimate release at BDBA < 1 E-7 /year
Design Status	Preliminary design completed; Key technologies are being demonstrated

The Power Conversion System Flow Diagram

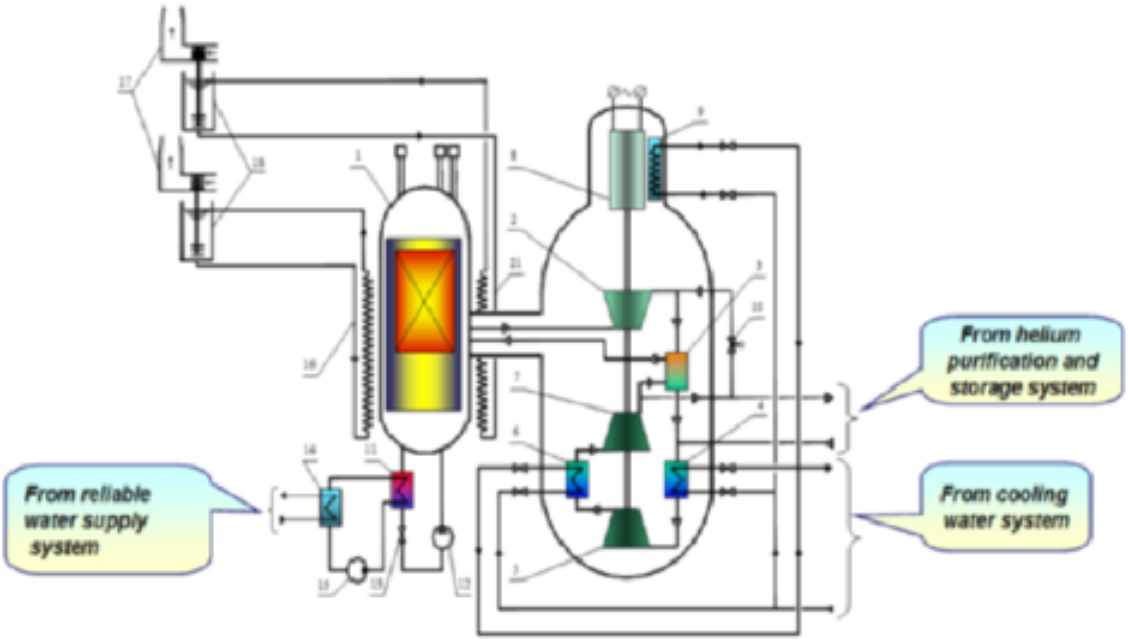


Figure 2: GT-MHR flow diagram (Reproduced courtesy of OKBM Afrikantov)

(1)-reactor, (2)-gas turbine, (3)-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), (21) – surface cooler of reactor cavity cooling system (RCCS), (17) – air ducts, (18) – heat exchanger with heat pipes

The Brayton power conversion with direct gas turbine is adopted that contains a gas turbine, an electric generator and gas compressors. This is shown in *Figure 2*. 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.

5. 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.

6. Plant Arrangement

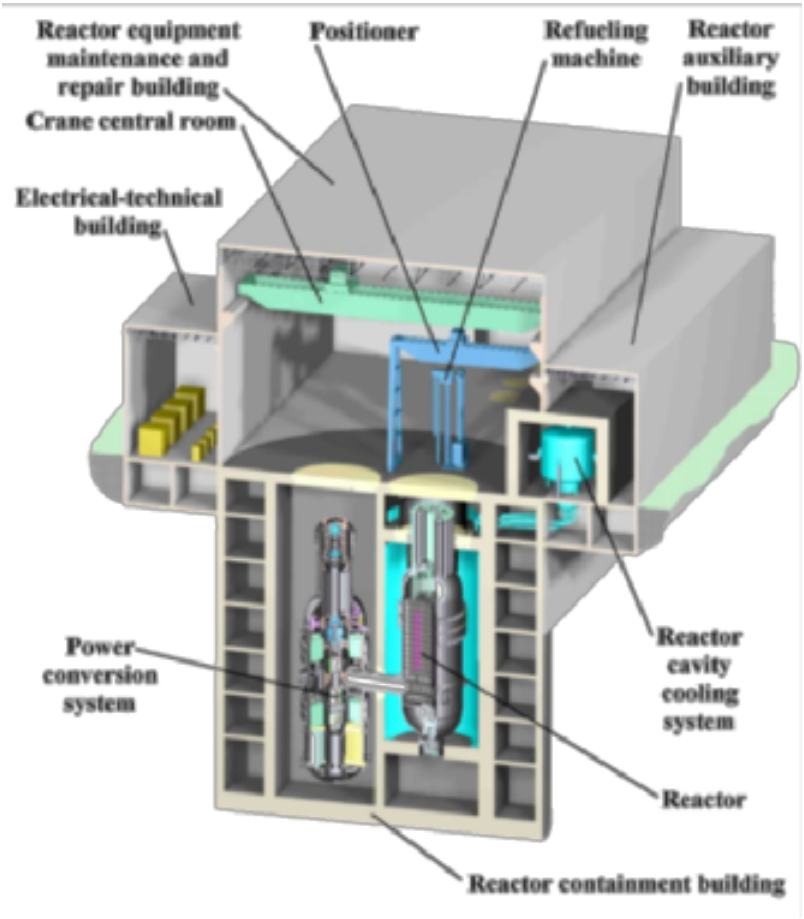


Figure 3: GT-MHR layout (Reproduced courtesy of OKBM Afrikantov)

7. Design and Licensing Status

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

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

1. 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.

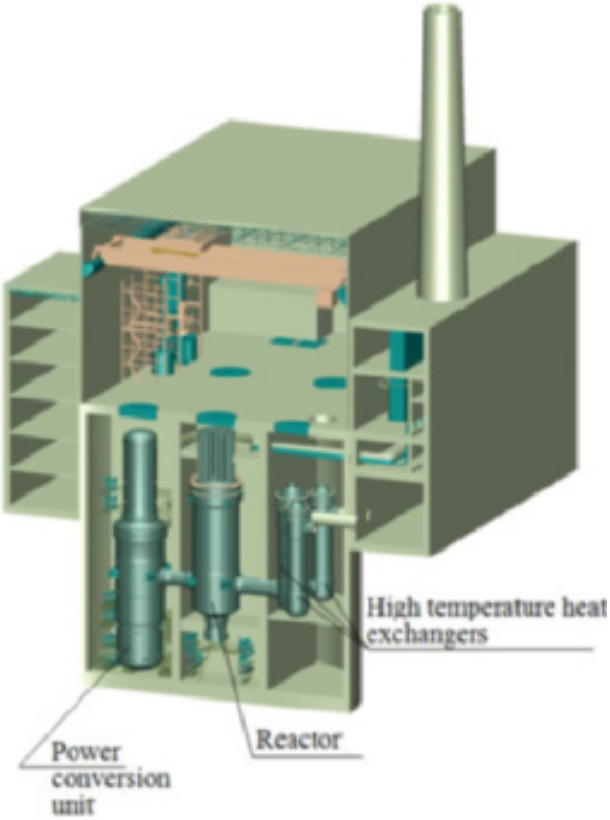


Figure 1: MHR-T RP layout – variant with steam methane reforming (Reproduced courtesy of OKBM Afrikantov)

2. Target Application

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.

3. Development Milestones

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

4. General Design Description

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.

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, H2, CO2, and CH4) in the course of a thermochemical reaction.

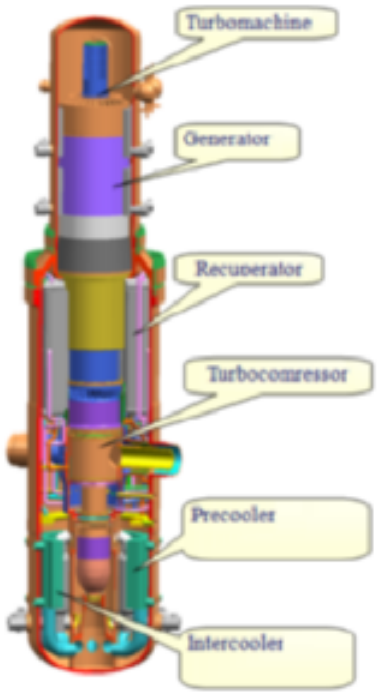
Reactor Core and Fuel Characteristics

Coated particle fuel is used. The fuel kernel (U 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°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.

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 (%)	80
Hydrogen or MHM production efficiency (gross), %	80
Hydrogen production (t/h)	4x12.5
Design life (years)	60
Plant footprint (m^2)	N/A
Coolant/moderator	Helium/graphite
Primary circulation	Forced circulation
System pressure (MPa)	7.5
Core inlet/exit temperatures (^oC)	578 / 950
Main reactivity control mechanism	Rod insertion
RPV height (m)	32.8
RPV diameter (m)	6.9
RPV or module weight (metric ton)	N/A
Configuration of reactor coolant system	Integral
Power conversion process	Direct closed Brayton cycle
Fuel type/assembly array	Hexagonal Prism graphite block of 0.36m coated particle fuel
Fuel assembly 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 (GWd/ton)	125
Fuel cycle (months)	30
Cogeneration capability	Yes
Approach to engineered safety systems	Hybrid (active + passive)
Number of safety trains	2
Refuelling outage (days)	25
Distinguishing features	Multi-module HTGR dedicated to hydrogen production / high temperature process heat application.
Modules per plant	4
Target construction duration (months)	48
Seismic design	8 points (MSK 64)
Core damage frequency (per reactor-year)	Core damage frequency not applicable to HTGRs BDBA frequency <1E-5/year Frequency of ultimate release at BDBA <1E-7 /year
Design Status	Conceptual design

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. 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.



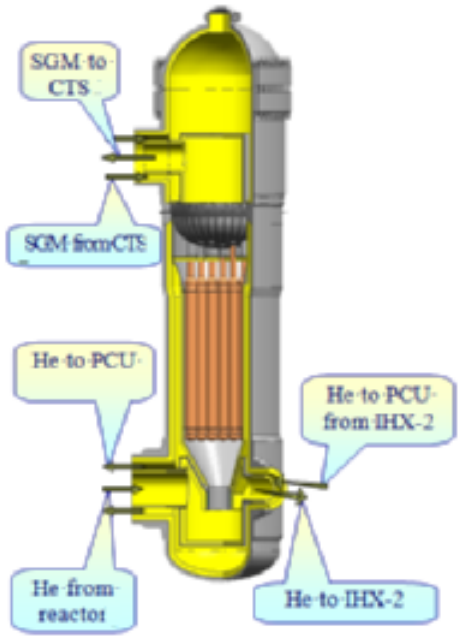


Figure 2a: Power conversion unit *Figure 2b: High temperature heat exchanger section*

(Reproduced courtesy of OKBM Afrikantov)

The turbomachine includes a generator and turbo-compressor 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%.

Reactor Coolant System

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.

5. 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 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.

Decay heat removal system

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.

6. Plant Arrangement

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.

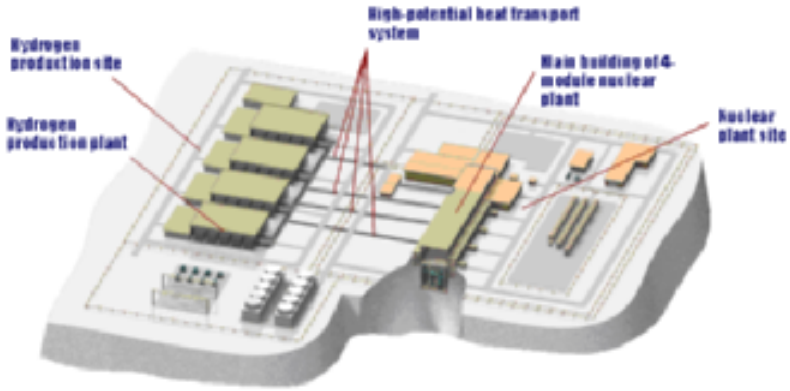


Figure 3: MHR-T site with hydrogen production complex
(Reproduced courtesy of OKBM Afrikantov)

7. Design and Licensing Status

Conceptual Design completed.

MHR-100 (OKBM Afrikantov, Russian Federation)

1. 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 MW(th), 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 MHR100 RP prototype with modular helium reactor and several power conversion layouts as sources of various power-industrial applications. Within the developmental work the following MHR100 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.

2. Target Application

The MHR-100 is intended for regional power generation and heat production in the Russian Federation. A single reactor unit design can be implemented in various plant configurations.

3. Development Milestones

2014	Conceptual design completed

4. General Design Description

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 power level (215 MW(th)) 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.

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 (%)	N/A
Design life (years)	N/A
Plant footprint (m^2)	N/A
Coolant/moderator	Helium / Graphite
Primary circulation	Forced circulation
System pressure (MPa)	4 – 5
Core inlet/exit temperatures (^oC)	490-553 / 795 – 950
Main reactivity control mechanism	Rod insertion
RPV height (m)	N/A
RPV diameter (m)	N/A
RPV or module weight (metric ton)	N/A
Configuration of reactor coolant system	Loop type
Power conversion process	Direct cycle Brayton, cogeneration / IHX
Fuel type/assembly array	Hexagonal Prism graphite block with coated particle fuel
Fuel assembly active length (m)	N/A
Number of fuel assemblies	N/A
Fuel enrichment (%)	LEU < 20% enriched
Fuel burnup (GWd/ton)	N/A
Fuel cycle (months)	N/A
Cogeneration capability	Yes
Approach to engineered safety systems	Hybrid (passive + active)
Number of safety trains	2
Refuelling outage (days)	25
Distinguishing features	A multipurpose reactor for electricity production, heat generation, and hydrogen generation; high-temperature heat supply to oil refinery plant
Modules per plant	Depending on deployment
Target construction duration (months)	48
Seismic design	8 points (MSK 64)
Core damage frequency (per reactor-year)	Core damage frequency not applicable to HTGRs BDBA frequency < 1E-5 /year Frequency of ultimate release at BDBA < 1E-7 /year
Design Status	Conceptual design

5. 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.

6. Design Variants based on the modular MHR-100

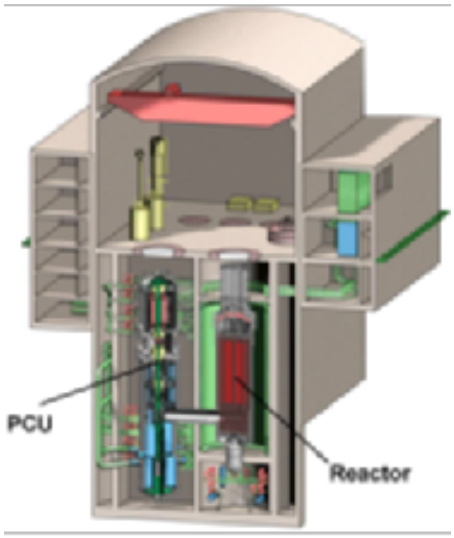


Figure 1: MHR-100 GT layout/options
(Reproduced courtesy of OKBM Afrikantov)

Major Technical Parameters of MHR-100 GT		
Parameter	**Power Mode**	**Cogeneration mode**
Reactor heat capacity (MW)	215	215
Net power generation efficiency (%)	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 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 UO2), high burnup and possibility to disposal the spent fuel blocks without additional reprocessing;
- 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.

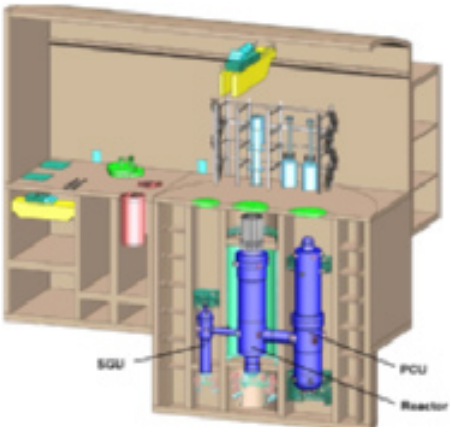


Figure 2: MHR-100 SE layout/option (Reproduced courtesy of OKBM Afrikantov)

Major Technical Parameters of MHR-100 SE	
Parameters	**Values**
Reactor heat capacity (MW)	215
Useful electric power of generator (MW)	87.1
Net power generation efficiency (%)	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/c)	6.46
Steam pressure at SG outlet (MPa)	4.82

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 pre-cooler 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 pre-cooler and intercooler by the cooling water system, then in dry fan cooling towers to atmospheric air.

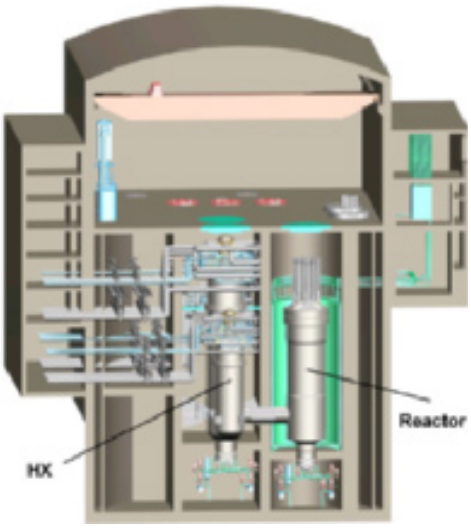


Figure 3: MHR-100 SMR layout/options (Reproduced courtesy of OKBM Afrikantov)

Major Technical Parameters of MHR-100 SMR	
Parameters	**Values**
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

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 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.

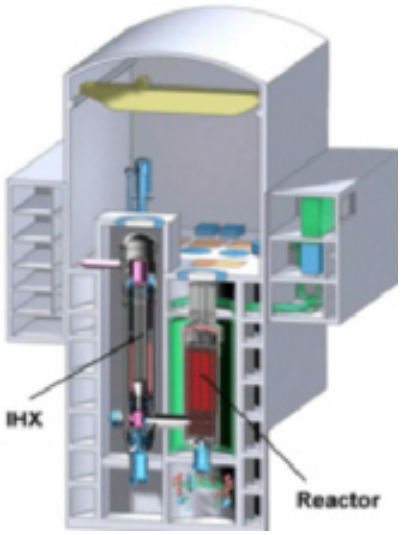


Figure 4: MHR-100 OR layout/options (Reproduced courtesy of OKBM Afrikantov)

Major Technical Parameters of MHR-100 OR	
Parameters	**Values**
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

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

1. 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. Various reactor concepts have been under development since 1996. Most of these designs are based on a direct Brayton cycle as this holds promise of higher efficiencies. The maximum achievable power levels for the reactor was increased in several design steps in order to reach a set target for installed cost/kW that would be roughly comparable to coal fired power when lifetime costs were evaluated. As a result the design of the reactor core evolved from the original base of 200 MW(th) adopted from the HTR-Module design to reach 400 MW(th) with an annular core, as shown in *Figure 1*. As a result of the world financial crisis in 2008 and short term funding constraints a rethink of the product priorities lead to a decision to concentrate on the electricity and process heat market with a single reactor product and thus a decision was made to use an indirect steam cycle. The direct cycle design was archived with a view to further progress this design when conditions (financial and technology development in materials for the direct cycle) improve.

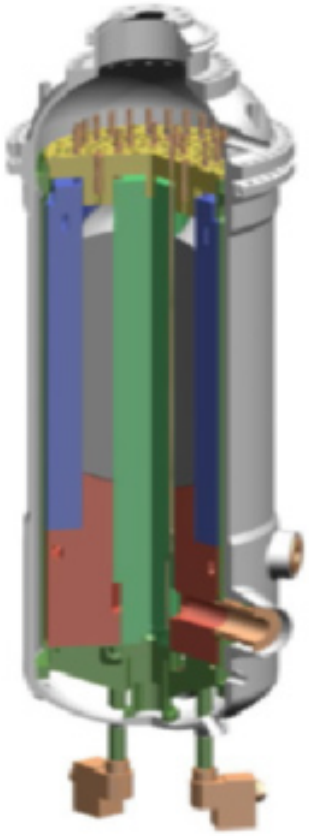


Figure 1: Reactor System Configuration of PBMR-400
(Reproduced courtesy of Pebble Bed Modular Reactor SOC Ltd.)

2. Target Application

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

3. 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

4. General Design Description

Design Philosophy

The PBMR-400 is a high-temperature helium-cooled, graphite moderated pebble bed reactor with a multi-pass fuelling scheme. 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.

The Brayton Cycle Power Conversion System

The Brayton power conversion with direct gas turbine is adopted. It is a closed cycle where the helium coolant is used to transport heat directly from the core to power turbine. The design incorporates a single shaft for the turbine, the compressors and the power generator. A recuperator is included. The layout is shown in *Figure 2* that also shows the pre- and inter-cooler as well as other auxiliary systems. From the reactor unit the hot helium enters directly to the turbine where energy is used to drive the shaft and therefore the electric generator and compressors. From the turbine the helium then passes consecutively through the low-pressure primary side of the recuperator, then the pre-cooler, the low pressure compressor, intercooler, high pressure compressor and then on to the high-pressure secondary side of the recuperator before re-entering the reactor unit. The direct gas cycle is attractive since it promises the benefits of simplification, with the potential of lowering the capital and operational costs. Due to the high outlet gas temperatures one will also expect a substantial increase in the overall system efficiency.

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
Core inlet/exit temperatures (^oC)	500 / 900
Main reactivity control mechanism	Negative temperature coefficient; Control rod insertion
RPV height (m)	30
RPV diameter (m)	6.2 (inner)
RPV or module weight (metric ton)	N/A
Configuration of reactor coolant system	Loop type
Power conversion process	Direct Brayton cycle
Fuel type/assembly array	Pebble bed with coated particle fuel
Fuel pebble diameter (cm)	6
Number of fuel assemblies	~452000 in core
Fuel enrichment (%)	9.6
Fuel burnup (GWd/ton)	92 (average discharge)
Fuel cycle (months)	N/A; Online / on-power refuelling
Cogeneration capability	No
Approach to engineered safety systems	Active
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.
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)
Target construction duration (months)	36 months (for the nth unit)
Seismic design	0.4g PGA for main power system design
Core damage frequency (per reactor-year)	Core damage frequency not applicable to HTGRs No off-site shelter or evacuation
Design Status	Preliminary design completed; Test facilities demonstration; Project stopped in 2010 and design in care and maintenance

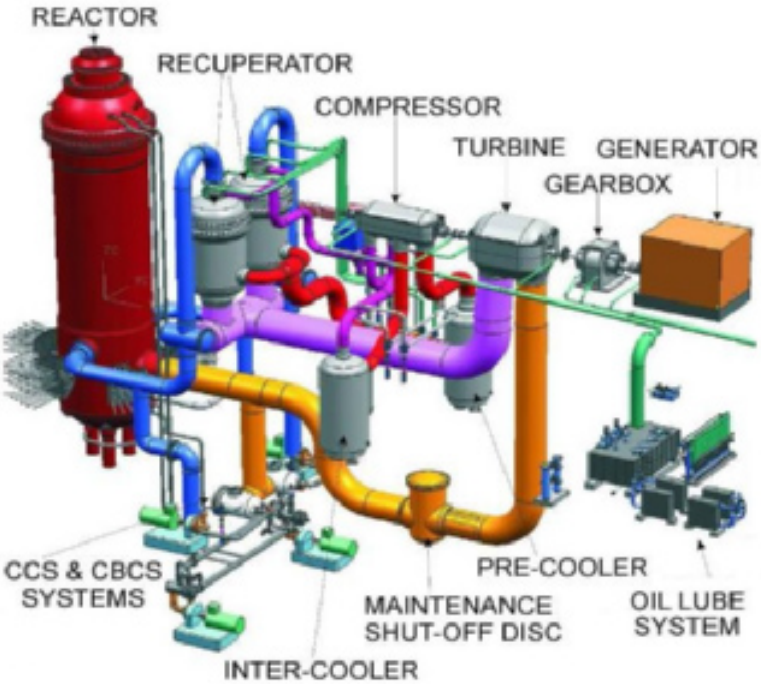


Figure 2: PBMR-400 power conversion unit components and layout
(Reproduced courtesy of Pebble Bed Modular Reactor SOC Ltd.)

The reactor power is adjusted by changes in the helium the mass flow rate in the power conversion unit. Since pressure changes will affect the helium density it is one mechanism to achieve this since the mass of the coolant will change accordingly. The helium inventory system is used and power control is performed in combination with bypass valves. Higher mass flows will remove more heat from the core eventually leading to an increase in the fission power through the natural feedback effects. Power reduction is achieved by removing gas from the circuit.

Reactor Core and Fuel Characteristics

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 ^{o}C) fulfil the design criteria. The core contains ~ 452000 fuel spheres or "pebbles" with a packing fraction of 0.61.

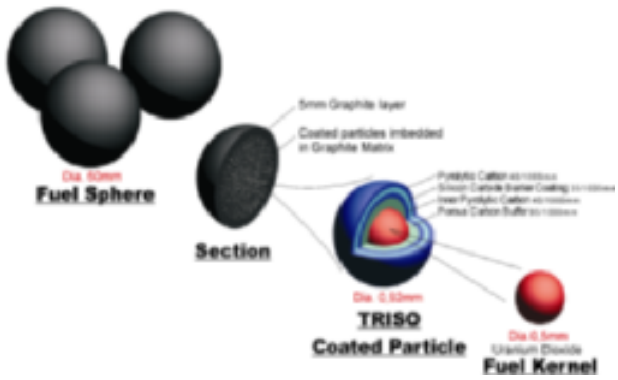


Figure 3: PBMR-400 coated fuel particle
(Reproduced courtesy of Pebble Bed Modular Reactor SOC Ltd.)

The fuelling scheme employed is the continuous on-line multi-pass method. Fresh fuel elements are added to the top of the reactor while used fuel pebbles are removed at the bottom to keep the reactor at full power. 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. Coated particle pebble fuel is used as shown in *Figure 3*. 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 15000 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.

Reactivity Control

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. 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%.

Heat Removal

The Reactor Cavity Cooling System (RCCS) is a means to remove residual heat passively for a defined time, and indefinitely with the use of an active system after refilling the cooling system. For this to work, the RPV and the core need to be long and slender. The belt region of the RPV is not insulated to allow heat radiation and convection to the water filled cavity cooler. In the event of the loss of active core cooling by the main circulation system, the cavity cooler and/or the building structural materials are able to limit the increase in fuel temperature in the most affected region of the core to below the allowable fuel temperature limit.

5. Safety Features

The safety philosophy for modular HTRs has been described a number of times in the past 30 years and has been adopted with a few modification by PBMR. It's basis is that an accident equivalent to severe core damage must be inherently impossible by limiting reactivity increases and ensuring that decay heat can be removed passively after a loss of coolant event. 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.

Barriers and Containment approach

The most important barriers to fission product release are the coatings of the fuel particles. High manufacturing requirements and strict quality control will ensure that the amount of free fissile or fertile material in the graphite matrix will be very low, and the percentage of fuel particles with missing or failed coatings will meet the requirements. This will ensure a low activity level in the primary circuit. A second barrier is provided by the Helium Pressure Boundary. This enclosure assures that only very small amounts of helium and nuclides circulating within it are released into the building during normal operation. A third barrier is the confinement building. This is that part of the reactor building which houses the primary coolant pressure boundary. The vented confinement is designed for very low leakage at low pressure, and to prevent damage to components important to safety, as well as to contain the build-up of higher activity gas in the delayed phase of a depressurisation event. Depending on the size of a pressure boundary break the system may be vented and then closed again with the released gas filtered as required.

6. Plant Arrangement

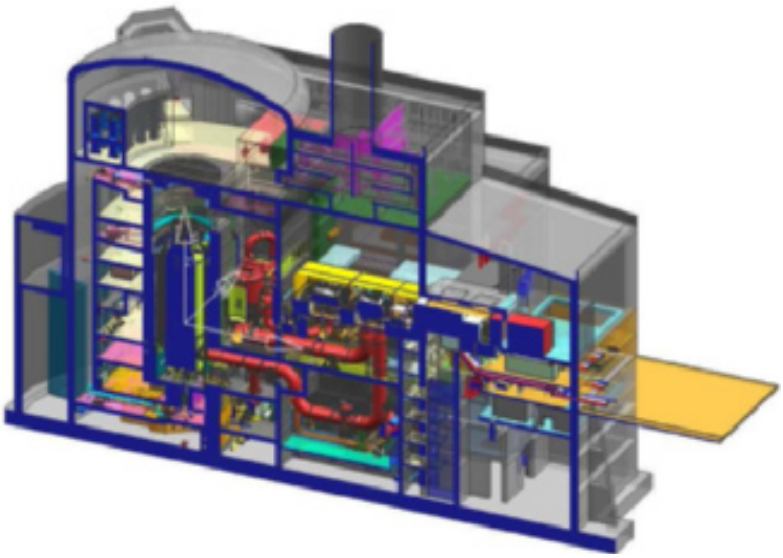


Figure 4: PBMR-400 building layout
(Reproduced courtesy of Pebble Bed Modular Reactor SOC Ltd.)

7. Design and Licensing Status

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

HTMR-100 SMR (Steenkampskraal Thorium Limited (STL), South Africa

1. Introduction

The HTMR-100 (High Temperature Modular Reactor) pebble bed reactor is a high temperature gas cooled reactor, graphite moderated and cooled by forced helium flow. The existing design of the module is to produce high quality steam which is coupled to a steam-turbine/generator system to produce 35MW of 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 an enhanced licensing process.

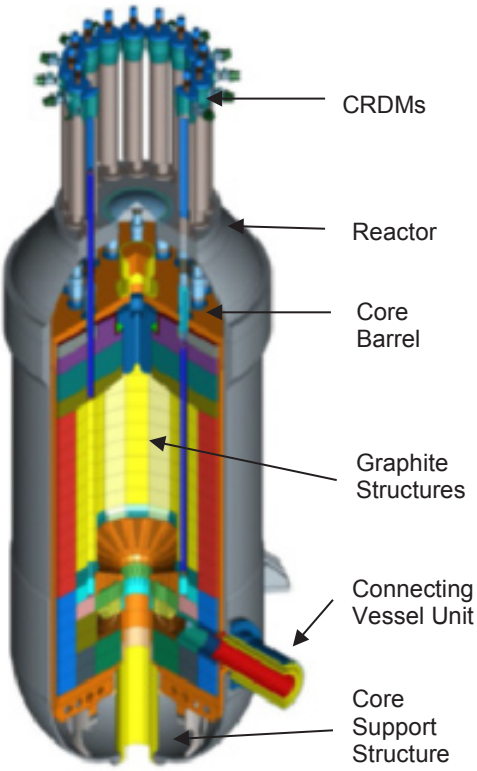


Figure 1: Reactor Configuration of the HTMR-100 (Reproduced courtesy of STL)

2. Target Applications

The HTMR-100 is capable of supplying electric power to any distribution grid and to standalone or isolated electricity users. It can be deployed as single modules or multi-module plants as well as 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.

3. Development Milestones

2012	Project started
2017	Preparation for Pre-license application
2018	Conceptual design finished

4. General Design Description

Design Philosophy

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

Reactor Core and Power Conversion Unit

The reactor unit consists of a steel pressure vessel, a steel core barrel, graphite reflector blocks, neutron absorber rods, rod guide tubes, drive mechanisms and in-vessel instrumentation. The vessel is designed for 4 MPa pressure. The graphite structure allows for differential expansion and volumetric changes due to temperature and neutron fluence induced distortion. This is done to keep the stresses low and minimize primary fluid bypass and leaks. The side, top and bottom reflector material is nuclear grade graphite.

The power conversion system uses a helical coil steam-generator unit supplying super heated steam to the turbine. The main system will be supplied as 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.

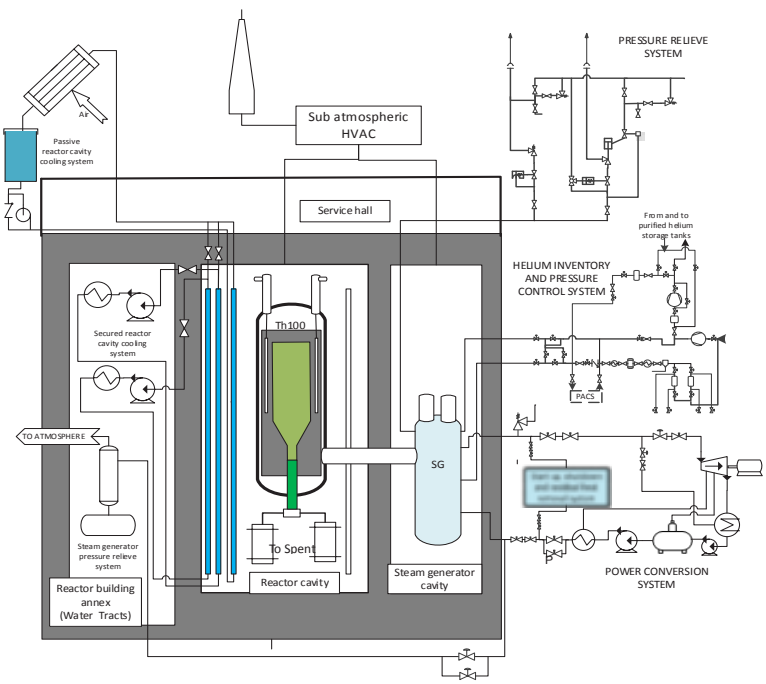


Figure 2: Power conversion and main support systems (Reproduced courtesy of STL)

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
Expected availability Factor	>95%
Design Life (years)	40 full power years
Plant Footprint (m^2)	5000 (Buildings only)
Coolant/moderator	Helium as coolant, graphite as moderator
Primary circulation	Forced He circulation
System pressure	4 MPa
Main Reactivity Control Mechanism	Absorber Rods in the reflector
RPV Height (m)	15.3
RPV Diameter (m)	5.9 on flange
Coolant Temperature, Core Outlet (°C)	750
Coolant Temperature, Core Inlet (°C)	250
Integral Design	Yes
Power Conversion Process	Indirect Rankine Cycle
High-Temp Process Heat	Yes
Low-Temp Process Heat	Yes
Co-Generation Capability	Yes
Design Configured for Process Heat Applications	Yes
Passive Safety Features	Yes
Active Safety Features	Only Residual Heat Removal backed up by passive systems
Fuel Type	TRISO particles in pebbles: LEU, Th/LEU, Th/HEU or Th/Pu
Fuel size	60 mm diameter
Number of Fuel units	~150000 pebbles ; about 125 pebbles/day throughput
Fuel Enrichment	Various – see fuel description
Fuel Burnup (GWd/ton HM)	80 to 90
Fuel Cycle	Continuous online fuel loading
Number of safety trains	No engineered safety train
Emergency active safety functions	None required
Residual heat removal systems	Passive and Active
Maintenance schedule (frequency and outage(days)	Every 6 years/30 days, 12 years/50days, mid-life 180days
Distinguishing features	No core meltdown, modular design , reduced construction time, no active ES systems, high quality steam , reactor upgradable to very high temperature, spent fuel in highly acceptable form, minimal production of tritium thus ideal for inland sites.
Modules per Plant	1 or multiples of 2 modules
Estimated Construction Schedule (months)	36 months , based on continuous manufacturing and construction
Seismic design	0.3 g for generic site. (0.5 g under consideration)
Core damage frequency	Slight damage with water ingress event with design base frequency < 1E-4 /year
Design Status	Advanced Concept Phase

Fuel characteristics and Supply

The fuel elements (FE) for the HTMR-100 are 60 mm diameter spheres 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 a fission product barrier in normal and accident operating 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. The following options have already been studied and show to be viable:

i. LEU : <10% Enrichment, (7-10g HM/sphere)
ii. Th/LEU: various options
    a. 50% LEU by mass 20% enrichment, 50% Th (10-12g HM/sphere)
    b. 25% LEU by mass 20% enrichment, 75% Th (16-20g HM/sphere)
iii. Th/HEU, 10% HEU by mass 93% enrichment, 90% Th by mass (10-12g HM/sphere)
iv. Th/Pu: 15% reactor grade Pu by mass, (12g HM/sphere)

A Fuel Qualification and Test programme will be conducted on the fuel prior to loading of the reactor. 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 more difficult to divert partially burnt fuel.

Fuel handling system

The Fuel Handling and Storage System (FHSS) is a support system to the Main Power System of the HTMR-100.

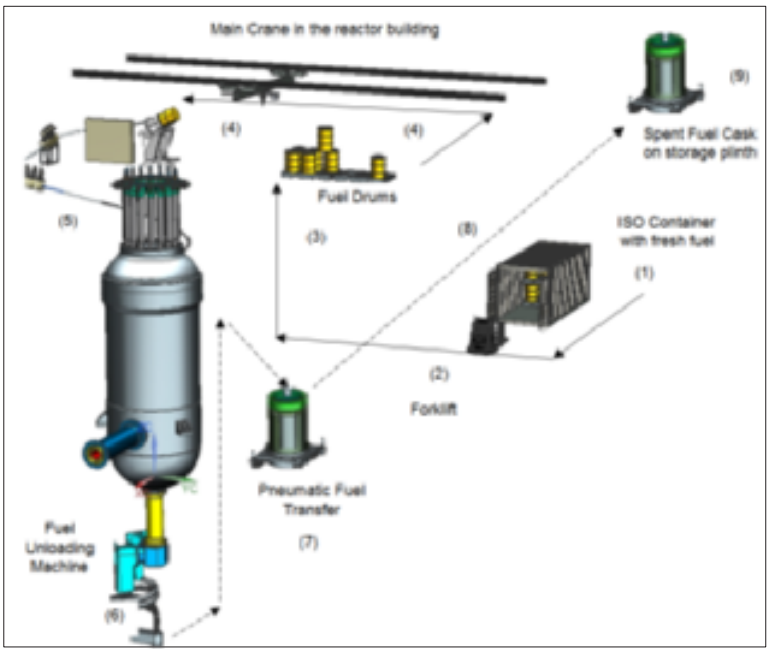


Figure 3: Fuel handling scheme (Reproduced courtesy of STL)

A six-month supply of fresh fuel is kept in the fresh fuel storage facility. New spherical fuel elements (fresh fuel) are loaded by the fuel loading machine into a charge lock. The charge lock is purged, filled with clean helium and pressurised to system pressure, before it is opened and fuel is gravity fed into the core cavity. The charge lock has a physical capacity for approximately one full-power day's fuel sphere inventory.

Spent fuel is removed from reactor pressure vessel by means of two fuel unloading machines (FUMs), connected to a single reactor defueling chute. Each FUM discharges spent fuel into a discharge lock. The discharge locks have sufficient capacity for a full power day of spent fuel elements.

Spent fuel, damaged in such a manner that gravity conveyance is compromised, is mechanically separated inside the FUM and accumulated in a dedicated scrap canister, located inside the FUM.

Provision is made in the design of the FHSS for the temporary fitment of activity measuring devices. These are to be used for the removal of the first core graphite spheres. Separate burn-up measuring devices can also be temporarily installed for core performance and calculational purposes when running fuel composition trials.

During decommissioning the FHSS extracts the last used fuel in the same manner as during normal operation.

Reactivity Control

Eighteen neutron absorber rods are provided in graphite sleeves inside the graphite side reflector blocks. The absorber rods can be operated independently as a group or as sub-groups, as required by the reactor operating control system. A control rod consists of several rod absorber material segments, pinned together to form articulating joints. The segments consist of sintered B_4C absorber material, sandwiched between an inner and an outer tube segment. The inner tube segment allows cooling helium gas to flow form the top down in the circular channels. Each rod is equipped with a position indicator which measures the position of the rod over its entire positioning range and with position indicators for the upper and lower limit positions.

Reactor Pressure Vessel and Internals

The Reactor Pressure Vessel (RPV) is constructed to the ASME III subsection NB code. It comprises of two main components:

- reactor vessel body
- vessel head which is bolted to the vessel body

The reactor vessel body consists of several forged ring-components circumferentially welded together.

The core structures consist of the metallic parts and the graphite structures. The function of these internal structures is to provide stable core geometry, neutron reflection, cold and hot gas channelling, fuel element flow, shielding, thermal insulation and support of the control and shutdown systems guide tubes and the neutron source. The functional design of the structural core internals is such that they are capable of withstanding the steady state and transient loadings during normal operation, anticipated operational occurrences and design basis accidents.

The shape and structure of the inner side reflector wall and the 30 ° angled core bottom permit uniform fuel element flow.

The loads borne by the ceramic internals are transferred to the steel core barrel and then to the reactor pressure vessel through metallic components such as the lower support structure and the core barrel axial and radial supports.

All areas of the core internals are designed for the service life of the reactor. Access for ceramic structure inspections can be done through the fuel loading channel and the reflector rod holes.

5. Plant Safety Features

Engineered Safety System Configuration and Approach

In principle the plant is designed to perform its safety functions without reliance on the automated plant control system, or the operator. The engineered safety system of the plant has no engineered safety systems in terms of active human or machine intervention to assure nuclear safety. 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. Provision for beyond design basis conditions is made. Beyond design basis scenarios include the non-functioning/non-insertion of all active control and shutdown systems. The reactor core characteristics e.g. small excess reactivity and strong negative reactivity coefficient with temperature will shut down the reactor and maintain a condition where no damage to the fuel, core structures and reactor vessel occurs. Excessive reactivity increase during water or water vapor ingress (increasing moderation) is prevented by designing the reactor for limited heavy metal content of the fuel.

Reactor Cooling Philosophy

The Reactor Cavity Cooling System (RCCS) removes heat radiated from the reactor towards the reactor cavity walls. It consists of welded membrane tubes arranged side-by-side on the inside of in the reactor cavity wall. Water is circulated through the tubes to form a cold wall. The RCCS is a passive system and consists of three independent cooling trains and is designed for all postulated design basis conditions.

Containment Function

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 building filter system) collapse.

6. Plant Safety and Operational Performances

The central consideration is the demand for high availability of process steam supply and/or electricity generation. To reduce or minimize the NSSS daily or weekly load changes of the reactor, the preference is to change the ratio between steam supply and electricity supply. Excess steam and/or electricity can be utilized in the desalination plants to provide water as a sellable commodity earning additional revenue. This allows the plant to operate virtually continually at full power very close to the plant availability. In order to maintain the plant safety and operational performance, specific areas are controlled for human occupation and protection.

7. Instrumentation and Control Systems

The Automation System (ATS) comprises that group of safety and non-safety I&C systems that provide automated protection, control, monitoring and human-system interfaces. The following three specific systems in the HTMR-100 system structure define control and instrumentation:

- Plant control, data and instrumentation system: Provides overall plant control and monitoring. It includes architecture which combines I&C systems and the control

room equipment.

- Equipment/Investment Protection System: Reduce the risk of losing critical or expensive components by maintaining the plant within its normal and safe operating envelope.
- Protection System: Initiates reactor SCRAM to protect against nuclear control failure or loss of primary coolant, etc.

The HTMR-100 ATS architecture implements modern digital I&C designs and equipment. The architecture of the overall Automation System is based on redundancy, diversity, separation and functional and physical independence. This concept includes four levels of defence-in-depth defence: (1) Distributed Control System or Control System; (2) Reactor Protection System; (3) Manual Shutdown System; and (4) Post-event Information System.

8. Site and Plant Layout

The HTMR-100 is configured to simplify the construction of the various buildings and to ensure easy installation of the reactor, steam generator and handling of spent fuel. The proposed plant layout aims to protect safety important functions while simplifying operational and maintenance tasks. The plant layout allows for the addition of multiple reactors (and associated equipment) in a compact multi-module fashion.

Reactor building

The reactor building contains the safety equipment that provides the necessary functions for the safe shutdown of the reactor under all design basis conditions. The reactor building is partially submerged below ground level such that the reactor and steam generator cavities are completely protected against postulated external threats. The depth can be further adapted to suit the geological conditions of the specific site to provide for the necessary level of seismic protection. The reactor building, electrical building and auxiliary buildings are connected by means of underground tunnels, providing protection for interlinked services and, it also ensures that spent fuel is never brought above ground level. Provision is made for the storage of all spent fuel produced during the operating life of the plant. The reactor building is seismically designed to withstand a design basis earthquake (DBE) and together with the spent fuel storage bunker, is the only safety related building structure of the HTMR-100.

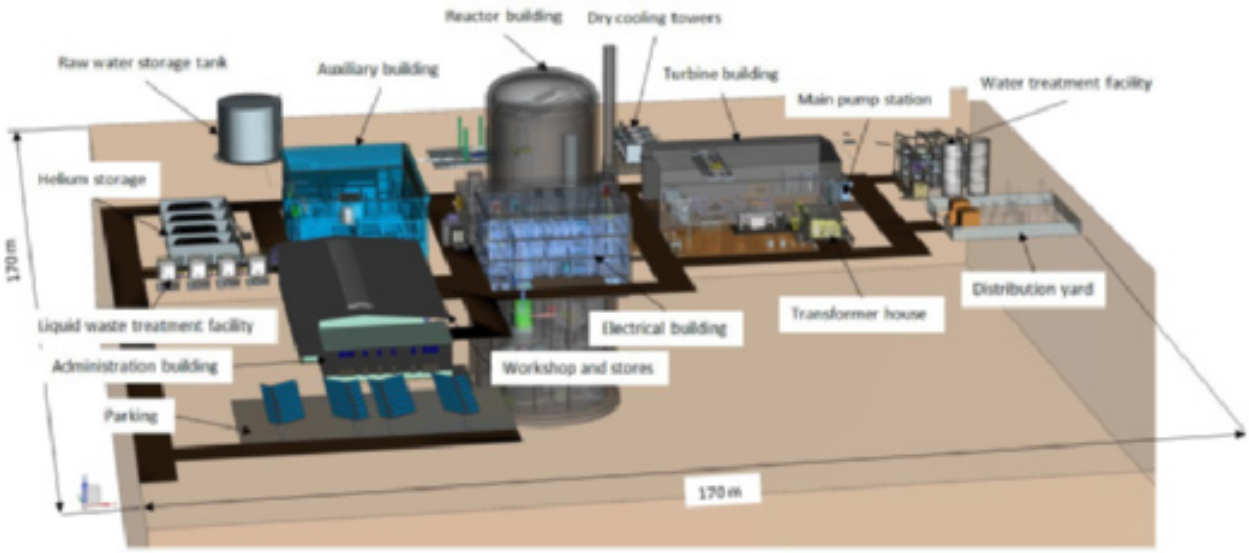


Figure 4: HTMR-100 Site and Plant Concept Layout (Reproduced courtesy of STL)

Electrical building
The EB houses the main control and computer rooms, primary and secondary plant security alarms rooms and provides the primary access facilities for the nuclear island and the energy conversion area. This centre also provides space for activities associated with plant administration and security services. The plant control, data, and instrumentation system control/display panels and computers are housed in the control room.

Turbine Generator Building
The Turbine Building provides the foundation and housing for the Power Conversion System, including other support systems such as the compressed air, water sampling, HVAC, Voltage Distribution Systems, permanent 11kVAC and 400VAC diesel generator sets and steam safety valves.

Balance of Plant
The balance of plant consists of the secondary systems and the tertiary systems. The Power Conversion System (PCS) has one set of isolation valves between itself and the Nuclear Steam Supply System (NSSS). The PCS and its major subsystems are of non-nuclear safety class thus conventional safety class and the product is of high quality industrial standard.

Electric Power Systems
The Electrical Power Supply System (EPSS) for the single module HTMR-100 plant supplies power to the plant safety and non-safety equipment for normal plant operation, start-up, normal shutdown as well as for accident mitigation and safe shutdown. The EPSS provides the following voltage levels: 230VAC, 400VAC, 11kVAC, 24/48VDC and 250VDC. The primary voltage of the Station Feeder Transformer is transmission grid dependent and thus site specific. The majority of plant loads are powered from the Main Power Supply System (MPSS) while all safety-related loads as well as non-safety related loads on the Nuclear Island (NI) portion of the EPSS are powered by the Reactor Power Supply System (RPSS). The Reactor Protection System (RPS) and some non-safety related loads (i.e. Primary Blowers 1 & 2, etc.) are powered from the RPSS. A portion of the RPSS circuit (inside the reactor building) is categorized as safety-related (Class1E).

9. Design and Licensing Status

Conceptual design is in an advanced stage. The core nucleonic, thermo-hydraulic and heat transfer analyses are being done to optimize the performance and verify the safety analysis. Nuclear Regulator engagement is planned for 2017 with the aim of commencing the pre-assessment for licensing in order to reach design certification status at the end of the Concept Phase.

10. Plant Economics

Since the HTMR-100 is mainly intended to service the needs of isolated geographical areas, it is not meant to compete economically with large (>1GW) reactors connected to an existing national grid. The HTR SMR comes into its own right when the various combinations of electricity, process heat and water desalination are considered and holds great promise for the long term economics of users. Preliminary capital and operating cost studies indicate that this flexibility of the HTMR-100 provides the most efficient energy to isolated communities and industries. It is well suited to deployment in developing countries with its affordability and Gen IV safety characteristics.

SC-HTGR (AREVA NP, USA)

1. Introduction

The SC-HTGR is a modular, graphite-moderated, helium-cooled, high temperature reactor with a nominal thermal power of 625 MWth and a nominal electric power capability of 272 MWe. It produces high temperature steam suitable for numerous applications including industrial process heat and high efficiency electricity generation. The safety profile of the SC-HTGR allows it to be collocated with industrial facilities that use high temperature steam. This can open a major new avenue for nuclear power use. The modular design allows plant size to be matched to a range of applications. The SC-HTGR concept builds on AREVA's past experience of HTGR projects, as well as on the development and design advances that have taken place in recent years for modular HTGRs. The overall configuration takes full advantage of the work performed on early modular HTGR concepts such as the MHTGR and the HTR-MODUL.

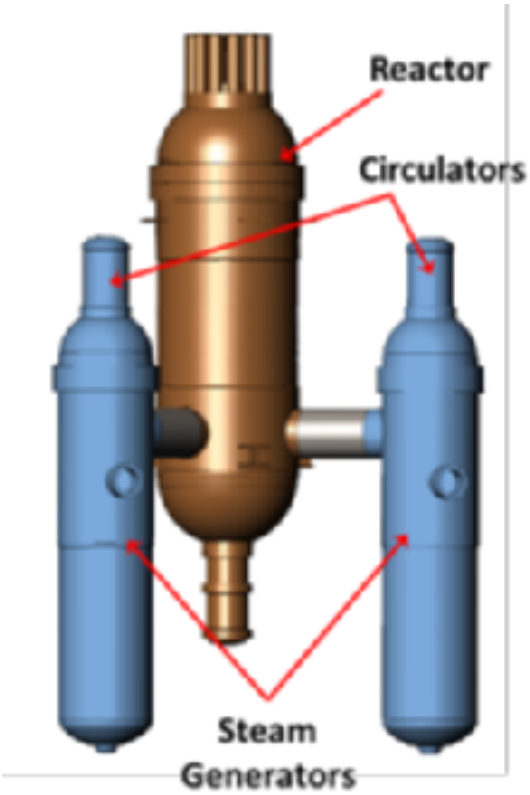


Figure 1: Nuclear Process Steam Supply System (Reproduced courtesy of AREVA)

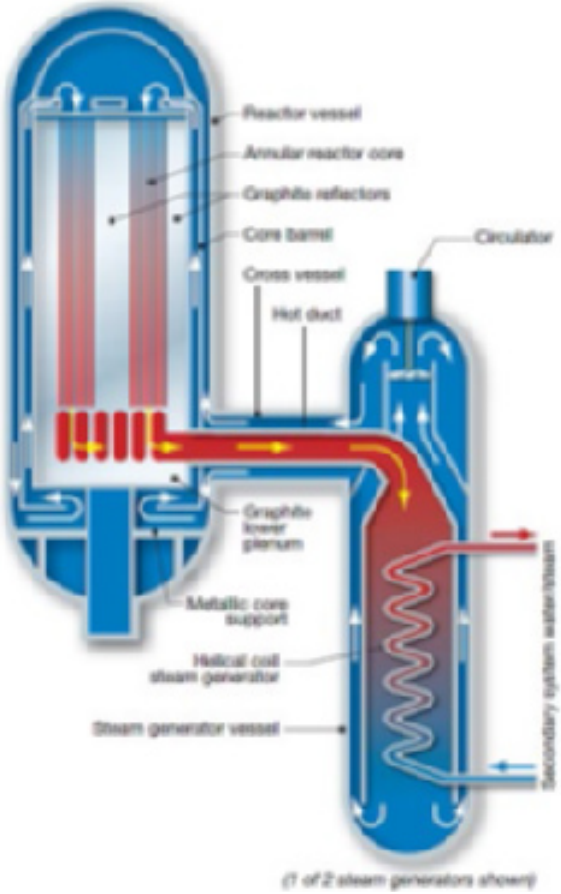


Figure 2: Primary Circuit Layout (Reproduced courtesy of AREVA)

2. Target Applications

The SC-HTGR produces high temperature steam suitable for numerous applications including industrial process heat and high efficiency electricity generation. The HTGR steam cycle 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. The steam cycle is also well suited to cogeneration of electricity and process heat.

3. Development Milestones

2009	Project Started
2011	Basic 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
2015	Confirmatory design studies in progress (precursor to full Conceptual Design)

4. General Design Description

Design Philosophy

The SC-HTGR is designed around proven helium-cooled and graphite moderated reactor technology, the heart of which is the TRISO coated fuel particle.

Reactor and Power Conversion Unit

The SC-HTGR reactor is a two-loop modular steam supply system. Each module consists of one reactor coupled to two steam generators. The steam generators are configured in parallel, each with a dedicated main circulator, as shown in *Figure 1*. For the reference plant steam cycle concept, the reactor power level is 625 MW(th).

A steel vessel system houses the entire primary circuit. The reactor vessel contains the reactor core, reactor internals, and control rods. Each steam generator is housed in a separate steam generator vessel. Each cross vessel contains a hot duct that channels hot gas from the reactor outlet to the steam generator inlet. Cool return gas flows in the outer annulus between the hot duct and the vessel wall. The entire vessel system inner surface is bathed in cool reactor inlet gas, so conventional LWR vessel material (SA-508/533) can be used. This is shown in *Figure 2*. The reactor vessel is partially insulated while the cross vessels and steam generator vessels are fully insulated.

Each steam generator is a once-through, helical coil tubular heat exchanger. Feed water enters the bottom of the steam generator and flows upward inside the tubes, while hot primary helium coolant flows downward over the tube bundle. The design of steam generators build upon the lessons learned at previously operated gas-cooled reactors (e.g., Fort St. Vrain, AVR, and THTR). Electric motor-powered main helium circulators provide the primary coolant flow. These variable speed circulators use submerged motors with active magnetic bearings for simple operation and high reliability.

The reactor inlet and outlet temperatures are 325°C and 750°C, respectively. These temperatures were selected primarily to support the desired steam outlet conditions for the target markets. These temperatures also allow the use of standard PWR vessel material, for the primary vessels without requiring separate cooling or special thermal protection.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology Developer	AREVA
Country of Origin	United States of America
Reactor Type	Prismatic Block HTGR
Electrical Capacity (MW(e))	272
Thermal Capacity (MW(th))	625
Minimum Availability Requirement (%)	>90
Design Life (years)	60
Coolant/Moderator	Helium / Graphite
Primary Circulation	2 loops; forced helium circulation – total of 282 kg/s
Steam Generator Power (MW(th))	315 (each)
Main Circulator Power (MW(e))	4 (each)
Net Electric Output in All Electric Mode (MW(e))	272
Primary System Pressure (MPa)	6
Main Reactivity Control Mechanism	Control rods (gravity insertion) Independent reserve shutdown system (gravity insertion) Large negative temperature coefficient
Reactor Vessel materials	SA508/533 (typical PWR vessel material)
Coolant Temperature, Core Outlet (°C)	750
Coolant Temperature, Core Inlet (°C)	325
Power Conversion Process	Rankine / Conventional steam cycle
Steam Generator Type	Once-through helical coil
Steam Generator Capacity	140.7 kg/s per loop (281.4 kg/s total) – 630 MW(th) total
Steam Conditions	566°C superheated steam at 16.7 MPa
Process Heat / Cogeneration Capability	Yes, High and low temperature or cogeneration
Design Configured for Process Heat Applications	Yes; 100% process heat utilization possible
Passive Safety Features	Passive decay heat removal; Passive shutdown capability
Active Safety Features	Active systems are not required to achieve safe state
Fuel Type/Assembly Array	UCO TRISO Particle fuel in hexagonal graphite blocks
Fuel Block Height (m)	0.8
Number of Fuel assemblies	Annular core, 102 column, 10 blocks / column,
Fuel Enrichment (%)	< 20
Fuel Burnup (GWd/ton)	< 17
Fuel Cycle (months)	½ of the core replaced every 18 months; 21 days outage
Emergency Safety Systems	Passive – No conventional ECCS systems
Residual Heat Removal Systems	Reactor Cavity Cooling System (Passive, water based); 2 natural circulation loops with 2.5MWth heat removal during accident and 7 days water storage capacity.
Distinguishing Features	Coated particle fuel; passive decay heat removal; passive safety; high temperature process steam
Modules per Plant	Variable (Reference design is four 625 MW(th) modules)
Design Status	Concept design

Fuel characteristics

The TRISO coated fuel particle consists of a uranium oxycarbide (UCO) fuel kernel surrounded by multiple ceramic coating layers that provide the primary fission product retention barrier under all design basis accident conditions. The total fuel inventory includes roughly 10 billion such particles per core. As shown in *Figure 3*, the particles are distributed in graphitic cylindrical compacts. Multiple compacts are contained within hexagonal nuclear grade graphite fuel blocks. The compacts are stacked in fuel holes drilled into the blocks.

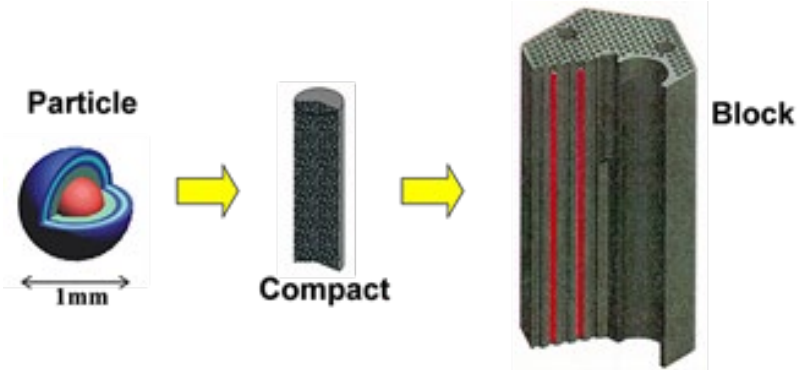


Figure 3: TRISO Fuel Particle, Compact and Block (Reproduced courtesy of AREVA)

Reactor Core Layout

The fuel blocks are configured into a 102 column annular core surrounded by graphite reflector elements as shown in *Figure 4*. The inner or central reflector also contains graphite reflector elements. Hence the basic core structure is entirely ceramic. This configuration maximizes the reactor's passive heat removal capability. The active core is 10 blocks high.

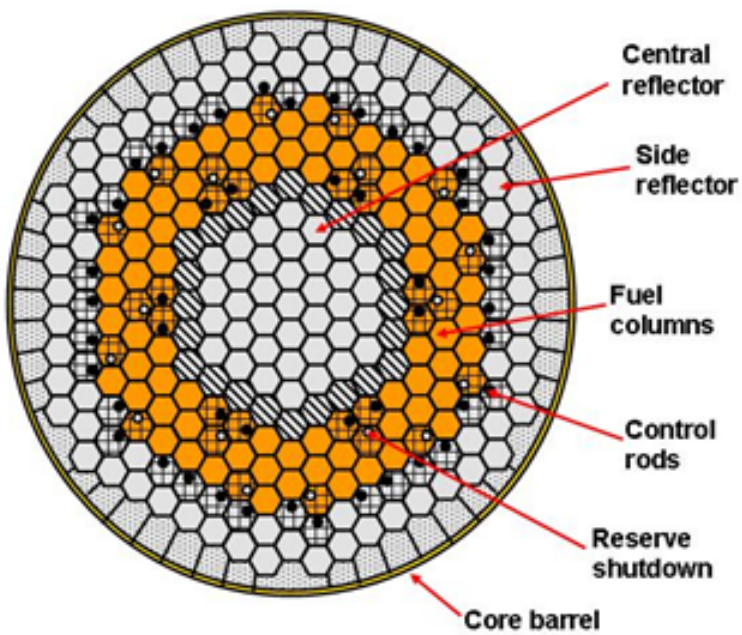


Figure 4: Annular Core Layout (Reproduced courtesy of AREVA)

Cycle Length and Fuel Management

The core cycle length for the SC-HTGR is between 420 and 540 effective full-power days. This has been confirmed for the initial core, using an initial core loading of 10.36 w/o U-235 enriched particles with a packing fraction of 0.289 for all fuel elements in the core, and for

reloads utilizing half-core replacement with fuel blocks having a 15.5 w/o U-235 enrichment and a packing fraction of 0.279 [1].

Control of local fuel power peaking and limiting of resulting peak fuel temperatures at critical locations within the fuel block will be accomplished through loading discrete burnable absorbers, variation of fuel packing fraction, and variation of fuel particle enrichment. This allows the optimization of the core power distribution in three dimensions and can also be used to support effective fuel utilization, proliferation resistance and waste reduction.

The high thermal efficiency and high fuel burnup of the SC-HTGR support sustainability for current once-through fuel cycles by minimizing spent fuel volume. The LEU once through fuel cycle requires about 6.8 MTHM/GWe-yr that equates to a natural uranium feedstock utilization of about 224 MT/GWe-yr. The SC-HTGR core design is also compatible with various more advanced fuel cycles employing fertile/fissile material conversion and recycle including Th/U, Th/Pu, Pu, and actinide fuel forms.

Reactivity Control

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.

5. Plant Safety Features

Description of safety concept

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 coated fuel particle. The coating system constitutes a micro-scale pressure vessel around each kernel that has been engineered to withstand extremely high temperatures without losing its ability to retain fission products 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 or not 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.

Security of the HTGR facility is provided both by the inherent invulnerability of the system to malicious acts and by the physical optimization of the facility structures to prevent unintentional access.

Engineered Safety System Configuration and Approach

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 various modular HTGR concepts including SC-HTGR throughout their licensing history.

Reactor Cooling Philosophy

The SC-HTGR has three heat removal systems. The two main cooling loops transfer heat to the secondary circuit during normal operation. When maintenance is being performed on the main cooling loops, a separate shutdown cooling system is available. This system uses a separate and independent circulator and heat exchanger located at the base of the reactor vessel. These systems also provide cooling during refueling and normal shutdown conditions as well as most Anticipated Events and DBEs.

If the above two active systems are unavailable, passive heat removal can be used. Heat from the core is conducted radially through the graphite reflectors to the core barrel and eventually to the reactor vessel. Heat is transferred from the vessel to the Reactor Cavity Cooling System (RCCS) by thermal radiation and natural convection. This heat removal path remains effective even if all primary coolant has been lost.

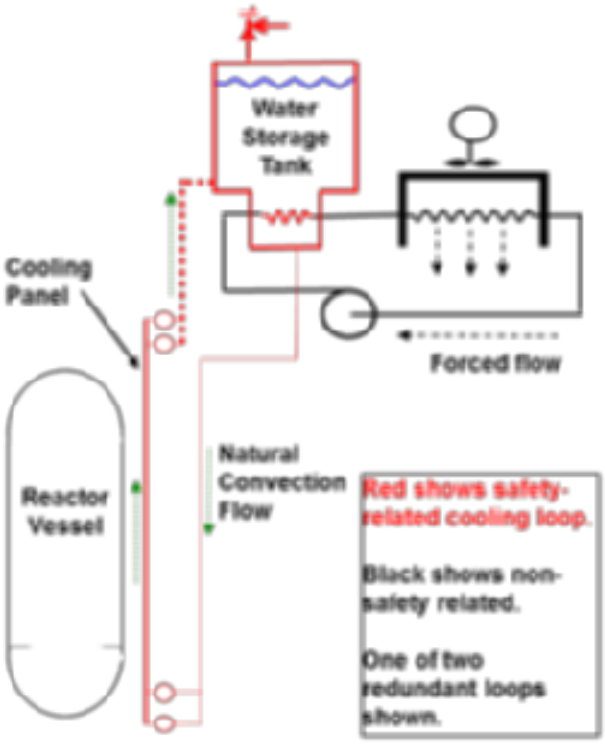


Figure 5: RCCS Flow Path (Reproduced courtesy of AREVA)

The RCCS, as shown in *Figure 5*, is a redundant natural circulation water-cooled system that maintains acceptable concrete temperatures in the reactor cavity during normal operation and Anticipated Events, and maintains acceptable fuel, vessel, and concrete temperatures during Design Basis Accidents. Each independent loop of the safety-related RCCS consists of heat

collecting panels in the cavity surrounding the reactor vessel connected by a natural circulation loop to a water storage tank. This loop uses natural circulation for all operating and accident conditions. A separate, non-safety-related active loop cools the tank during normal operation. The water in the tank provides the required thermal capacity for a minimum of 7 days of continued cooling during accidents when the active system may not be available.

Containment Function

The radionuclides containment function in the SC-HTGR is performed primarily by the TRISO fuel coatings. The graphite core structures, primary coolant boundary, and reactor building provide supplemental containment capability. The SC-HTGR reactor building uses a vented confinement system. The building provides supplemental fission product retention in the event of an accident. However, a pressure retaining building such as a light water reactor containment building is not necessary or technically appropriate due to the excellent fission product retention performance of the fuel even under extreme accident conditions.

6. Plant Operational Performance

The thermal-hydraulic performance of the SC-HTGR reactor core is linked directly to the nuclear core design to ensure that the power distribution results in acceptable fuel temperatures for the whole core for all operating modes over the life of the core.

The nominal electricity generation performance of the SC-HTGR system has been evaluated [2], taking into account preliminary efficiency estimates for the helium circulators, feedwater pumps, turbine, generator and other plant electrical loads. The net electrical output from each 625 MW(th) reactor module is 272 MW(e), for a net efficiency of 43.5%. In addition to nominal plant performance, the performance of the SC-HTGR has already been evaluated for hot arid locations where dry cooling was assumed to be required. Results of this evaluation indicate that a net electrical generation output of 239 MW(e) is achievable, for a corresponding efficiency of 38.2%.

Steam system equipment can be configured in a variety of ways depending on the specific needs of the user facility requirements for high temperature process steam, low temperature process steam, and electricity. The secondary and tertiary system of the plant can be easily customized for each energy user application without affecting the reactor module configuration or safety case. The steam cycle plant also has excellent load following characteristics for flexible power operation and grid stabilization for micro-grids with non-dispatchable sources. Reactor module power level and steam production can be increased or decreased on demand and relatively easily. Systems can also shift energy between electricity generation and heat supply dynamically as real-time load and market conditions vary, all while keeping reactor power constant. This provides the maximum utilization of the HTGR nuclear heat source.

The overall reliability of an industrial energy delivery system using SC-HTGR as a key component was evaluated for several potential energy plant configurations [3]. The results of the study demonstrated that the extreme availability and reliability requirements of a process heat plant can be met with HTGRs as the primary heat source.

7. Plant Arrangement

Each reactor module is located in a separate reactor building. The standard configuration uses a fully embedded below grade reactor building design as shown in *Figure 6*. This provides structural design advantages and superior protection from external hazards. An alternative partially embedded configuration can be used for sites where a fully embedded structure is not appropriate. The primary functions of the reactor building are to support the NSSS primary circuit components and to protect the system from external hazards.

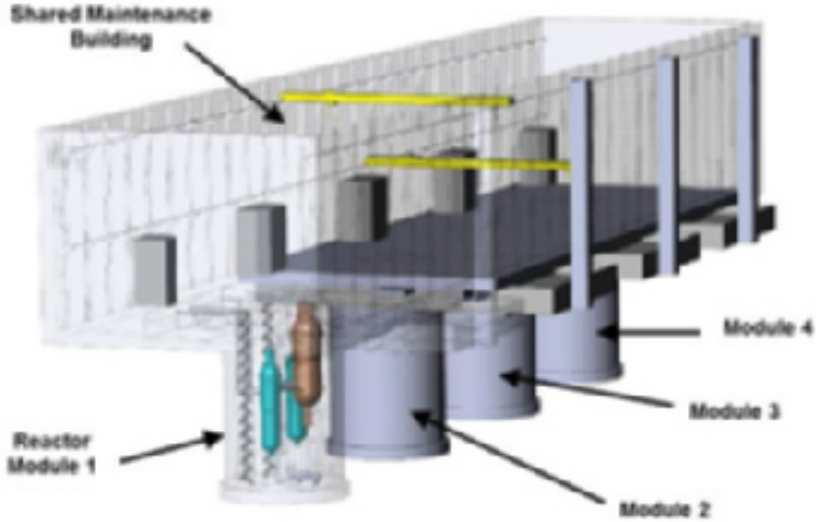


Figure 6: Standard Reactor Building Configuration (Reproduced courtesy of AREVA)

Multiple reactor modules can be grouped together on a single plant site. A typical plant layout might have four reactor modules, although the specific number of modules in an actual plant, and the timing of construction of each individual module, will depend on the nature of the application and the customer's needs. Reactor modules share auxiliary, support and maintenance systems, but safety systems, including the Reactor Cavity Cooling System and Reactor Protection System, are independent.

8. Design and Licensing Status

A concept design and preparatory work for pre-licensing application has been completed.

9. Plant Economics

A business and commercialization plan has been developed as part of the NGNP Industry Alliance (www.ngnpalliance.org) SC-HTGR commercialization activities.

[1] INEEL/EXT-03-00870, Rev. 1, "NGNP Preliminary Point Design – Results of the Initial Neutronics and Thermal-Hydraulic Assessments During FY-03", September 2003.

[2] Lommers, L. J., et. al., "SC-HTGR Performance Impact for Arid Sites", Proceedings of the HTR 2014, Weihai, China, Paper HTR2014-21345, October 2014.

[3] Herd, E., Lommers, L.J., and Southworth, F., "Impact of demand load size on strategies for reliable process heat supply", Nuclear Engineering and Design, Vol 251, October 2012.

Xe-100 (X-energy, United States of America)

1. 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 80000 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.

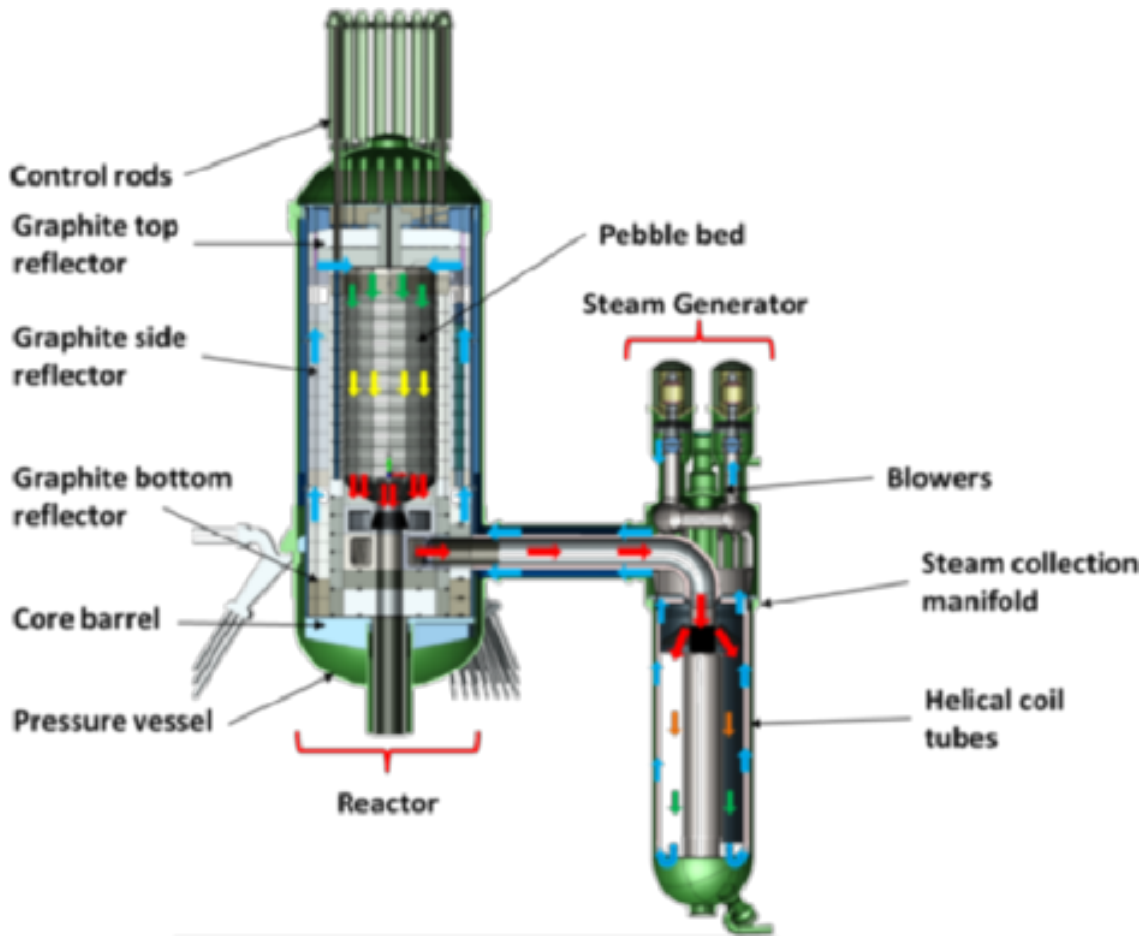


Figure 1: Xe-100 Reactor System Configuration (Reproduced courtesy of X-energy)

2. Target Application

The Xe-100 is intended for electricity production suitable for small or isolated grids. It can also provide super-heated steam for co-generation, 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.

3. Development Milestones

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

4. General Design Description

Design Philosophy

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.

Reactor Core

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 137Cs.

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 (NC/NA) 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 remains 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.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	X-energy LLC
Country of origin	United States of America
Reactor type	Modular High Temperature Gas-cooled Reactor
Electrical capacity (MW(e))	35
Thermal capacity (MW(th))	100
Expected capacity factor (%)	> 95
Design life (years)	40
Plant footprint (m^2)	100000 (4 reactor modules with one turbine); 160000 (8 reactor modules with two turbines)
Coolant/moderator	Helium / Graphite
Primary circulation	Forced confection; He flow Top → Bottom through core
System pressure (MPa)	4
Core inlet/exit temperatures (^oC)	260 / 750
Main reactivity control mechanism	Bank 1 (9 Rods) + Bank 2 (9 Rods)
RPV height (m)	15
RPV diameter (m)	5.15
RPV or module weight (metric ton)	–
Configuration of reactor coolant system	Loop type
Power conversion process	Indirect Rankine Cycle
Fuel type/assembly array	Pebbles
Core volume (m^3):	27.7
Number of fuel assemblies	–
Fuel enrichment (%)	10.61
Fuel burnup (GWd/ton)	79.7
Fuel cycle (months)	On-line refuelling
Cogeneration capability	Yes
Approach to engineered safety systems	Hybrid (Passive and Active)
Number of safety trains	–
Refuelling outage (days)	N/A
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
Target construction duration (months)	24 – 36
Seismic design	0.15g operating limit & 0.3g safe shutdown
Core damage frequency (per reactor-year)	No core melt possible
Design Status	Conceptual Design development

5. Safety 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.

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.

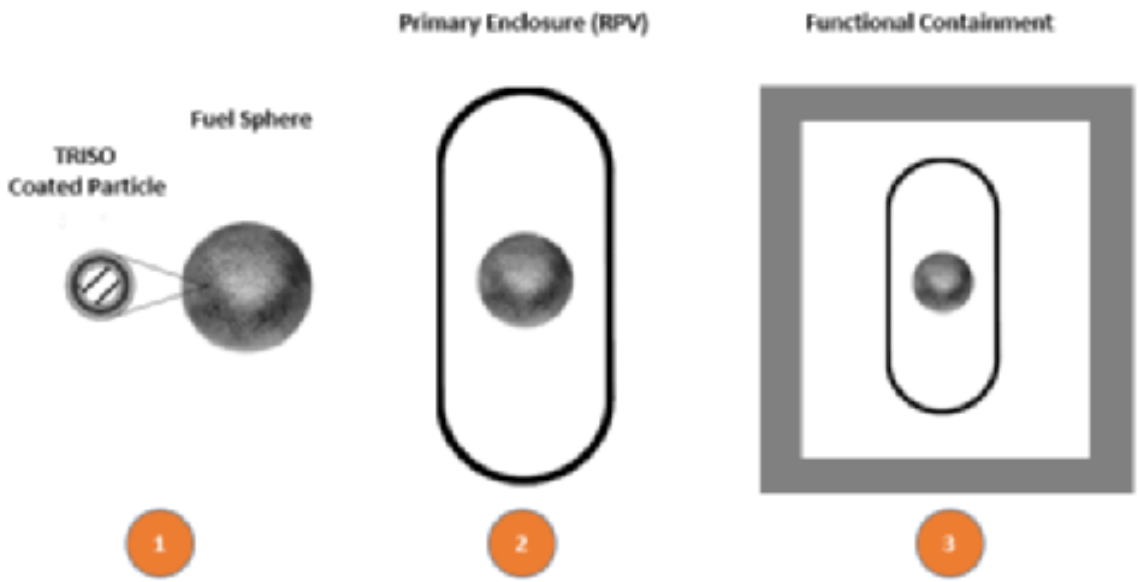


Figure 2: Concept of Independent Fission Product Barriers
(Reproduced courtesy of X-energy)

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.

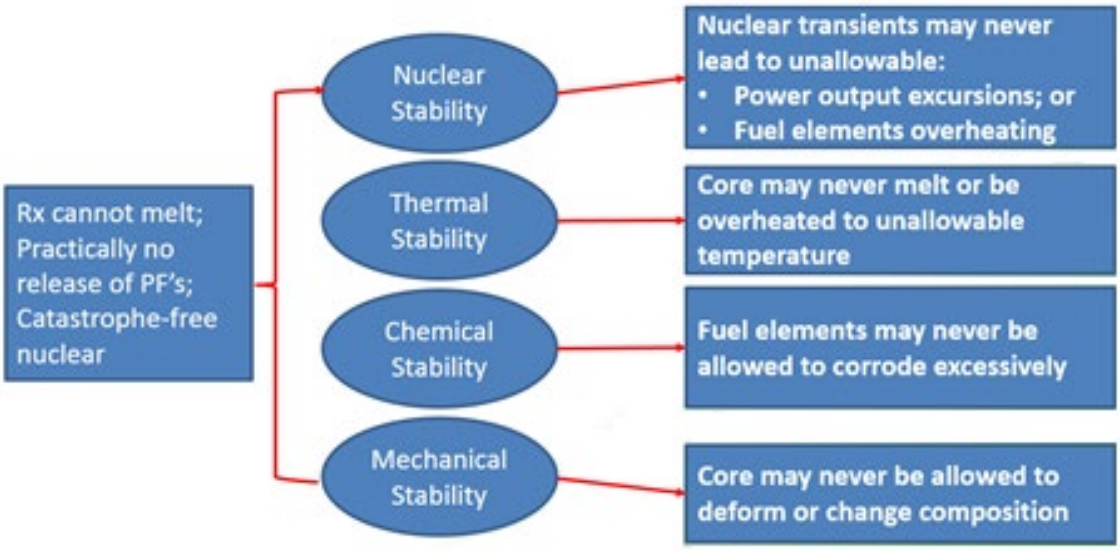


Figure 3: Stability Design: 4 Principles (Reproduced courtesy of X-energy)

6. Plant Arrangement

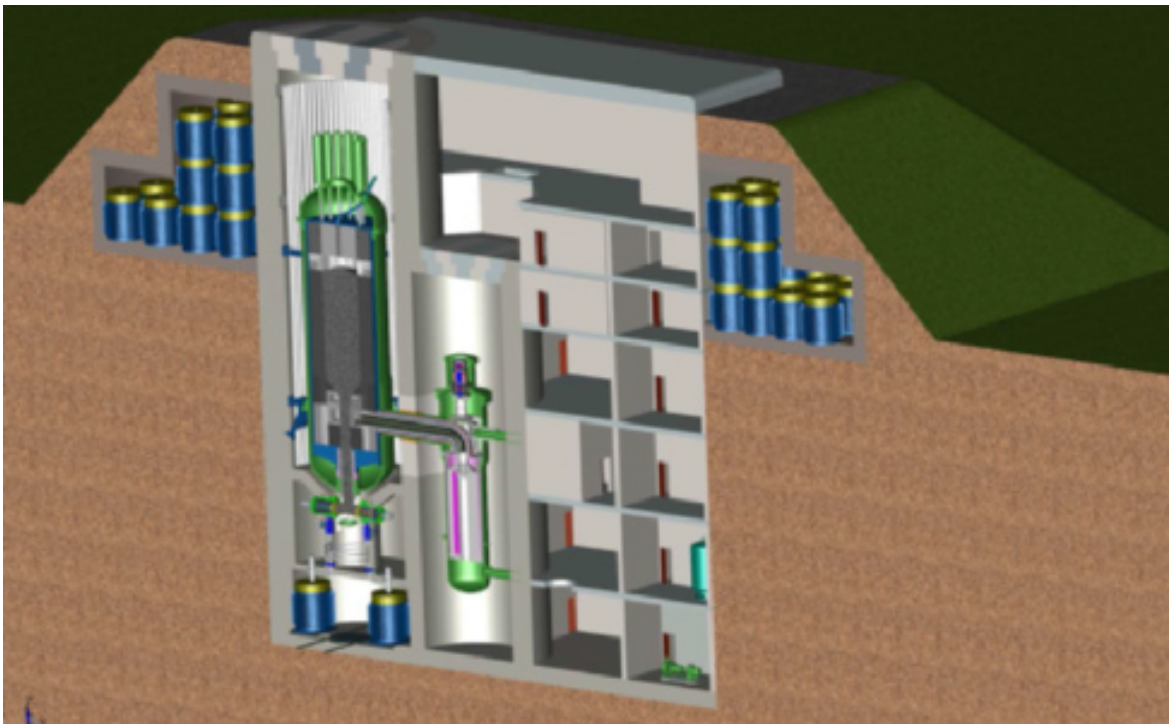


Figure 4: Conceptual Reactor Module Design (Reproduced courtesy of X-energy)

7. Design and Licensing Status

Conceptual design development and preparation for pre-licensing application.

FAST NEUTRON SPECTRUM SMALL MODULAR REACTORS

LEADIR-PS (Northern Nuclear Industries Incorporated, Canada)

1. Introduction

LEADIR-PS stands for Lead-cooled Integral Reactor-Passively Safe, a small nuclear reactor being developed by the Northern Nuclear Industries Incorporated (N^2I^2) intended to provide diverse electricity and process heat demands of Canada. LEADIR-PS plants under development are LEADIR-PS30 (30 MW(th), up to 11 MW(e)), LEADIR-PS100 (100 MW(th), up to 39 MW(e), and LEADIR-PS300 (300 MW(th), up to 120 MW(e)). Derivatives of these plants include LEADIR-PS Six-Pack, which features six LEADIR-PS300 modules that serve a common Turbine Generator, deliveries 720 MW(e), and LEADIR-PS30+ that incorporates thermal energy storage in the LEADIR-PS30 Balance of Plant to facilitate electrical outputs of from 5.5 MW(e) to 15 MW(e) over a 24 hour cycle while maintaining 100% reactor power. A simplistic illustration of the LEADIR-PS reactor assembly is presented in *Figure1*.

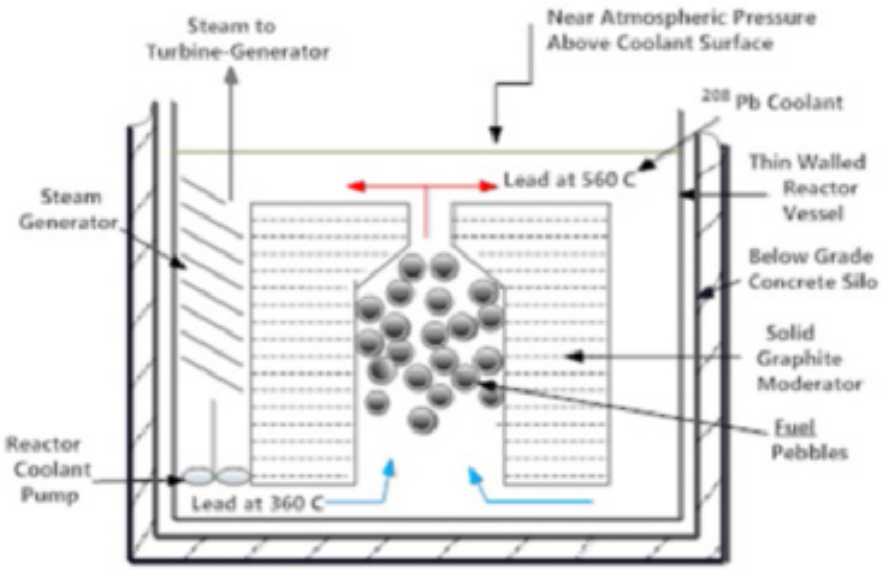


Figure 1: Reactor system configuration of LEADIR-PS (Reproduced courtesy of N^2I^2)

2. Target Applications

LEADIR-PS100 can serve a wide range of electricity, process heat, and district heating demands. By providing 12 MPa steam at 370°C, LEADIR-PS units can severe a diverse range of process heat, combined heat and electricity, and electricity demands. Energy utilization in process heat and combined heat and electricity (CHE) applications can exceed 95% and can be optimized to suit specific applications. Combined heat and electricity applications can employ a back pressure turbine that exhausts to a process heat application. As the process application serves as the condenser, electrify is generated without any heat rejection to the environment. Extraction steam capability can be incorporated into the turbine design to facilitate the delivery of steam to process users at different pressures. The application configuration of LEADIR-PS is shown in *Figure 2*.

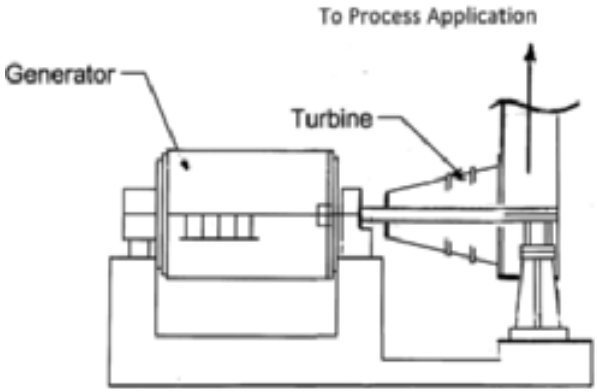


Figure 2: LEADIR-PS application configuration (Reproduced courtesy of N^2I^2)

3. General Design Description

Design Philosophy

LEADIR-PS is a molten lead (Pb_{208}) cooled, which in conjunction with the design configuration precludes a loss of coolant accident (LOCA) and the loss of decay heat removal capability. Pb_{208} does not react with graphite, water, or air. LEADIR-PS is graphite moderated, which via the negative reactivity temperature coefficient, assures inherent reactor shutdown at elevated temperatures.

Reactor Coolant System

LEADER-PS is a pool type reactor that operates with near atmospheric pressure above the Pb_{208} reactor coolant surface, thereby avoiding high a pressure vessel. It is an integral reactor with steam generators (SGs) and reactor coolant pumps located within the Pb_{208} coolant pool thereby avoiding all high pressure reactor coolant system piping. The primary cooling system is based on forced circulation during normal operation. The system has natural circulation capability for emergency condition. The primary reactor cooling system (PRCS) is contained within the reactor module assembly. The primary reactor cooling system includes two steam generation modules (SGMs), each consisting of two vertical once through SGs and one variable speed reactor coolant pump (RCP). Water/steam flows upward within the SG tubes and the Pb_{208} coolant flows downward outside of the Steam Generator tubes.

Steam Generator

The SG shell comprises two overlapping concentric cylinders, the SG upper shell and the SG lower shell that facilitate their relative vertical motion. The resulting 'floating' feedwater tubesheet is free to move vertically within the SGM support structure. This configuration prevents stress in the SG tubes due to thermal expansion and contraction of the SG tubes. Molten Pb_{208} has excellent lubrication qualities. The Pb_{208} coolant flows downward through the SGs where the heat of fission is transferred to ordinary water within the SG tubes to generate superheated steam. On exiting the SGM, the Pb_{208} coolant enters the suction of a reactor coolant pump (RCP). On being discharged from the RCP the Pb_{208} flows downward.

Nuclear Steam Supply System

The main steam system delivers steam from the SGs to the turbine and to in-plant and external process steam users. The main steam system includes the condenser steam discharge valves, the atmospheric steam discharge valves and the main safety relief valves. In some applications, two or more LEADIR-PS modules may be configured to supply steam to a single turbine-generator. In this case, dedicated feedwater control is provided to each LEADIR-PS unit.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology Developer	Northern Nuclear Industries Incorporated
Country of origin	Canada
Reactor type	Liquid metal cooled fast reactor
Electrical capacity (MW(e))	39
Thermal capacity (MW(th))	100
Expected Availability Factor (%)	–
Design Life (years)	60
Plant Footprint (m^2)	–
Coolant/moderator	Pb_{208} / Graphite
Primary circulation	Forced circulation
Number of reactor coolant pump	2 (Vertical Centrifugal)
System pressure (MPa)	12
Core inlet/exit temperatures (^oC)	360 / 560
Main Reactivity Control Mechanism	Control rods, on-power fuelling system
RPV Height (m)	10.2
RPV Diameter (m)	3.2
Module Weight (metric ton)	–
Configuration of Reactor Coolant System	Integrated
Power Conversion Process	Indirect Rankine Cycle
Fuel Type/Assembly Array	TRISO coated particle
Fuel material	U_{235} oxide
Fuel pebble diameter (mm)	60
Number of Pebble Bed cells	3
Number of fuel pebbles in core	94,500
Fuel Enrichment (%)	9.8
Fuel Burnup (GWd/ton)	86
Fuel Cycle (months)	–
Breeding ratio	–
Cogeneration Capability	Yes
Approach to Engineered Safety Systems	Passive
Number of safety trains	–
Refuelling Outage (days)	–
Distinguishing features	Eliminates the need of pressure vessels; low boiling point coolants; Multi-purpose reactor for electricity, process heat, district heating and combined heat and electricity.
Modules per plant	1
Target Construction Duration (months)	24
Seismic design	–
Predicted core damage frequency (per reactor year)	–
Design Status	Conceptual design

Reactor Core

LEADIR-PS utilizes a core design based on the Pebble Bed High Temperature Gas Reactor (HTGR), which was developed in Germany in the 1960s for the AVR-15 and THTR300, and which has been adopted by China for their HTGR. LEADIR-PS subdivides the core into three (3) pebble bed cells, as shown in *Figure 3*. Each pebble bed cell, essentially a small reactor consisting of a cylindrical vertical cavity in the graphite reactor structure occupied by Fuel Pebbles, has an output of 34 MW(th). The number of pebble bed cells depends on the reactor output (one in LEADIR-PS30, three in LEADIR-PS100, and nine in LEADIR-S300). The fuel pebble contains 16,500 TRISO particles. There are 32,500 Fuel Pebbles in a pebble bed cell. The graphite moderator, reflector and core barrel constitute the reactor core assembly (RCA). The reactor core assembly is housed within the Pb_{208} filled primary reactor vessel (PRV). Pb_{208} has a relatively low thermal neutron capture cross-section which contributes to excellent fuel utilization. The fuel pebbles and graphite moderator/reflector blocks are buoyant in the Pb_{208} coolant.

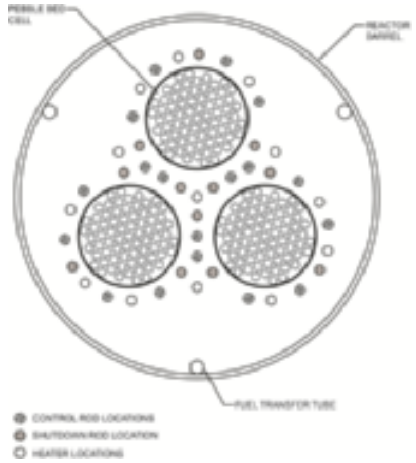


Figure 3: Reactor core plan section (Reproduced courtesy of N^2I^2)

LEADIR-PS utilizes 0.92 mm diameter TRISO fuel particles (0.6 mm diameter uranium oxide kernel with multiple coatings), with an enrichment of 9.8% to obtain sufficient uranium mass in the reactor core and to provide for high burnup. New fuel pebbles contain TRISO particles with a small amount of burnable neutron absorber that to serves to limit the heat generated in the pebble during their initial period in the reactor core. The coatings retain fission products under all potential conditions thereby precluding a radioactive release to the environment. The interlocking graphite moderator and reflector blocks forms the principal structure of the reactor core and define the cylindrical pebble bed cells. The graphite block structure is designed to accommodate thermal expansion and to minimize the creation of passageways resulting from graphite block distortion caused by radiation. The graphite structure is a vertical cylinder and together with the reactor barrel (RB) constitutes the RCA.

Reactivity Control

The three pebble bed cells are neutronically closely coupled. The small LEADIR-PS100 core is stable in all modes. Reactor control takes advantage of the strong negative reactivity temperature coefficient of the graphite reflector/moderator and the high heat capacity of the core. The on-power fuelling system is the primary mechanism for reactor reactivity control. Operational reactor control is provided by the reactor control system (RCS). The RCS monitors and controls all functions that are related to reactivity control, including the on-power refuelling system, except for the reactor shutdown system. The fully automatic RCS utilizes dual redundant computers that incorporate numerous checks and safeguards.

Reactor Pressure Vessel and Internals

The primary reactor vessel (PRV) is housed within the secondary reactor vessel (SRV). The annular space between the PRV and the SRV, called the annulus, is filled with lead enriched in neutron absorbing lead isotopes. This lead consists of the 'tails' from the Pb_{208} isotope separation process plus ordinary lead. A vertical cylindrical divider within the annulus, with openings at the top and bottom, divides the annulus into two annular cylindrical sections. The integral pool reactor configuration in combination with the high boiling point Pb_{208} coolant avoids the need for a pressure vessel and high pressure primary RCS piping, thereby eliminating the potential for high energy pressure vessel or piping rupture.

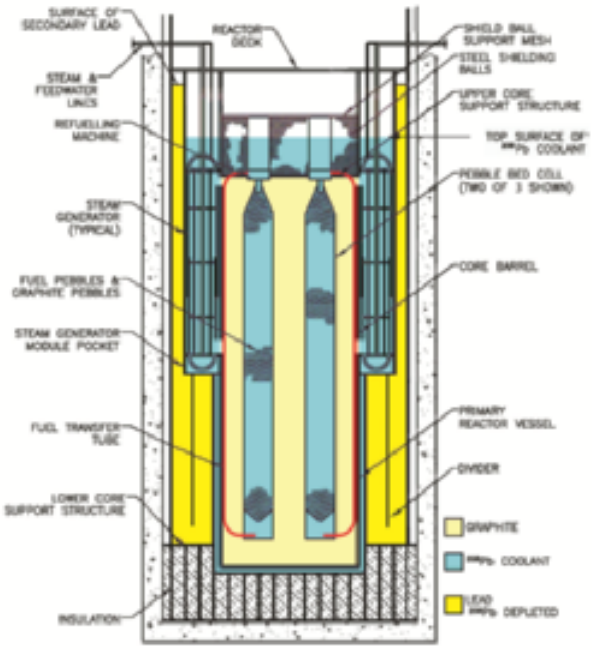


Figure 4: Simplified LEADIR-PS100 Reactor Module Assembly Elevation Section (Reproduced courtesy of N^2I^2)

4. Safety Features

LEADIR-PS provides an unprecedented level of safety, which includes:

- No potential for a reactivity transient without shutdown/scram;
- No potential for a LOCA, loss of heat sink and containment failure;
- No potential for pressure vessel or high pressure RCS failure;
- No potential for hydrogen generation, by coolant disassociation or zirconium water reaction;
- No potential safety impact by incompetent or malicious operator/maintenance staff or third party via the control system or component degradation;
- No requirement for operator action, valve repositioning or any power source;
- No requirement for any above ground structures or components to be functional.

LEADIR-PS is located in a below grade silo for protection from external events and to provide an assured long term heat sink. The reactor is inherently and passively safe requiring no power sources, no operating systems, not repositioning of valves and no operator actions to assure safety. LEADIR-PS remains inherently safe in the event that all above grade systems and structures are lost, and in the unlikely event of a fire above the reactor assembly.

Engineered Safety System Approach and Configuration
The passive shutdown system incorporates shutdown rods (SDRs) that drop by gravity with spring assist when power is removed from electromechanical clutches. SDRs deployment by the reactor shutdown system is backed up by passive initiation by the passive shutdown initiation system (PSIS) that removes power from the clutches when a specific core outlet temperature is reached. Inherent shutdown capability is provided by the negative temperature coefficient of the graphite, thereby assuring reactor shutdown in the event that both the reactor shutdown system and PSIS fail when needed. The high structural capability of graphite at high temperature (maximum strength at 2500°C) assures core integrity under all postulated conditions. LEADIR-PS safety, including inherent shutdown and passive decay heat removal capability is maintained even if the above ground portion of the Reactor Protective Structure is extensively damaged or demolished.

Decay Heat Removal System
After the reactor is shutdown, when the normal decay heat removal mechanism utilizing the secondary system is not operable by any reason, core submergence and natural circulation of the molten lead coolant is assured. As lead in the annulus is solid and would remain in place during normal reactor operation even if the secondary reactor vessel is not present this situation is extremely unlikely. Passive decay heat removal is assured for all credible events for an indefinite period of time without the need to reposition valves or other devices and without any power or any operator actions. Although provided, natural circulation of molten lead coolant is not required to assure decay heat removal.

Containment System
The coatings retain fission products under all potential conditions thereby precluding a radioactive release to the environment. No reactor containment is required. The location of the reactor assembly below grade and submerged in lead in combination with the passive decay heat removal capabilities makes the core highly resistant to external events. Additional protection is afforded by the above ground reactor protective structure.

5. Plant safety and Operational Performances

Heat removal is via the SGs utilizing the main feedwater system (MFWS) and the main condenser (MC) during most periods. Turbine steam bypass to the condensers enables continued reactor operation at 100% full power for up to 5 minutes and indefinite operation at up to 60% powers in the event that the Turbine-Generator is unavailable. LEADIR-PS100 operating characteristics can be summarized as:

- Black start capability independent of the electricity grid;
- 100% power operation without an electricity grid connection for pure process heat applications and operation at up to 100% power as the only electricity supply to the electricity grid;
- The reactor can maintain sustained operation at any power level between 10% and 100% of full power;
- Reactor control is normally in the reactor following load (electricity and/or process heat) mode;
- The power design increase rate for the reactor and reactor systems is 0.2 percent of full power per second between 10% and 100% power;
- The power reduction design rate for the reactor systems is 0.2 percent of full power from 100% power to 60% power and 0.05 percent per second below 60% power;
- More rapid load reductions (process heat or electricity) results in steam bypass to the condenser until reactor power drops to the required level.

6. Plant Arrangement

The compact layout of a single module LEADIR-PS100 is illustrated in *Figure 5*. The principal structures are the reactor protective structure (RPS), the turbine building (TB), the electrical and control building (E&C), the radioactive storage building (RSB), the auxiliary services building (ASB), and the access corridor. The E&C building structure is integrated with that of the TB and is separated from the TB by a waterproof and fireproof wall. The E&C building is seismically and tornado qualified to in accordance with site conditions. Additional onsite facilities include the switch yard, and the pump house if water cooled Main Condensers is used. An accommodations building (AB) which provides sleeping, kitchen, lunch, and relaxation facilities for the maintenance crews when on site is provided for LEADIR-PS100 units at remote or isolated locations. The layout of plants with multiple LEADIR-PS modules is similar to that of the single module plant, but employs an integrated RPS design, and may, depending on project specific energy demands, have a common turbine-generator serving more than one LEADIR-PS100 module.

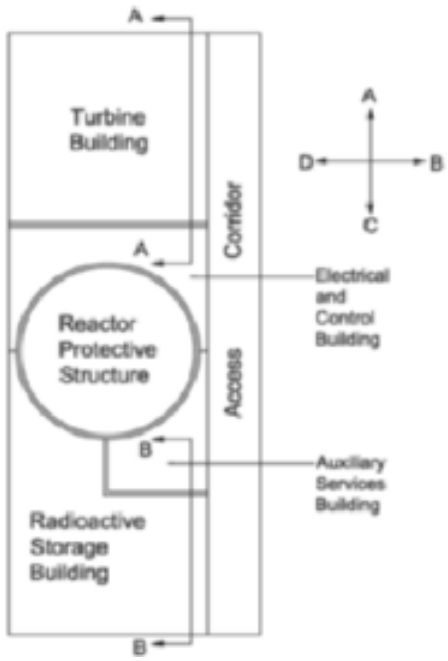


Figure 5: LEADIR-PS general plant arrangement with a single unit (Reproduced courtesy of N^2I^2)

Reactor Building

The reactor building and fuel storage area are equipped with a full monitoring system with closed circuit monitoring system (CCTV). *Figure 5* shows the overview of the reactor building, auxiliary building, and compound building arrangement of LEADIR-PS.

Balance of Plant:

Turbine Generator Building

The TB is a three level structure located to the 'A' side of the E&C building. The bottom level of the TB, located below grade, houses the plant water systems, the services boiler (SB), and the main condenser (MC) if a water cooled main condenser is used. The second level of the TB accommodates the turbine-generator (TG), the TG support systems including the lube oil system, the generator hydrogen system, the diesel generators and mechanical shop area.

Electric Power Systems

These systems include the main generator, main transformer, unit auxiliary transformers, stand-by auxiliary transformers, diesel generators, and batteries. The electrical systems, including the class 1E and non-class 1E, are based on a two train approach. The electrical services (ES) provide power for lighting, heating, and the operation of tools and machinery to all plant buildings as required. Power to ESs is provided by the Group A and Group B power supplies. The ESs is connected to their power supply only when a maintenance team is present. Power supply to the ESs can be established remotely by the central operations facility.

7. Plant Economics

The advanced delivery technologies of LEADIR-PS100 facilitate short and secure construction schedules. For LEADIR barge delivery subsequent to the first unit, the construction schedule is 24 months. This includes 15 months for the fabrication of the LEADIR barge, 1 month for the delivery of the LEADIR barge and 8 months for the completion of site work and commissioning. Site preparation and the construction of site support facilities proceeds in parallel with LEADIR barge fabrication. The schedule for fully modularized on-site construction, for LEADIR-PS units subsequent, to the first unit, is 30 months.

4S (Toshiba Corporation, Japan)

1. Introduction

The 4S (super-safe, small and simple) is a small sodium-cooled pool-type fast reactor with metal fuel. Being developed as distributed energy source for multi-purpose applications, the 4S offers two outputs of 30 MW(th) or 10 MW(e) and 135 MW(th) or 50 MW(e), respectively. These energy outputs are selected from the demand analyses. The 4S is not a breeder reactor since the blanket fuel, usually consisting of depleted uranium located around the core to absorb leakage neutrons from the core to achieve breeding of fissile materials, is not present in its basic design. 4S reactor cores are designed to have a lifetime of 30 years for the 30 MW(th) core and 10 years for the 135 MW(th) core. The movable reflector surrounding the core gradually moves, compensating for the burnup reactivity loss over the core lifetime. The plant electric power can be controlled by the water–steam system, which makes the reactor applicable for a load follow operation mode.

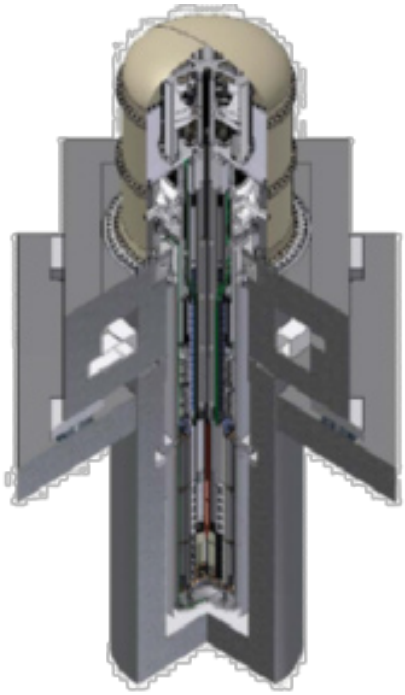


Figure 1: Reactor system configuration (Reproduced courtesy of Toshiba Corporation)

2. Target Application

The 4S is designed for electricity supply to remote areas, mining sites as well as for non-electric applications. The plant can be configured to deliver hydrogen and oxygen using the process of high temperature electrolysis. This process can be performed without producing environmentally disadvantageous by-products, such as carbon dioxide. Two kinds of systems for non-electric applications can be incorporated in the 4S;

- Seawater desalination system: the 50MW(e) 4S plant can produce fresh water at a rate of 168000 m^3/day;
- Hydrogen and oxygen production system: hydrogen production rate for the 10MW(e) and 50 MW(e) 4S is 3000 Nm^3/h and 15000 Nm^3/h respectively.

Combinations of these systems and the turbine generator system as balance of plant (BOP),

including the capacity of each system, would be determined to meet the actual needs at any particular site.

3. Development Milestones

2007	Licensing activity for the 4S design initiated with the U.S. Nuclear Regulatory Commission (U.S.NRC)
2008	Completion of 4 times public meetings as pre-application review with U.S.NRC
2013	Completion of submitting 14 technical reports to U.S.NRC

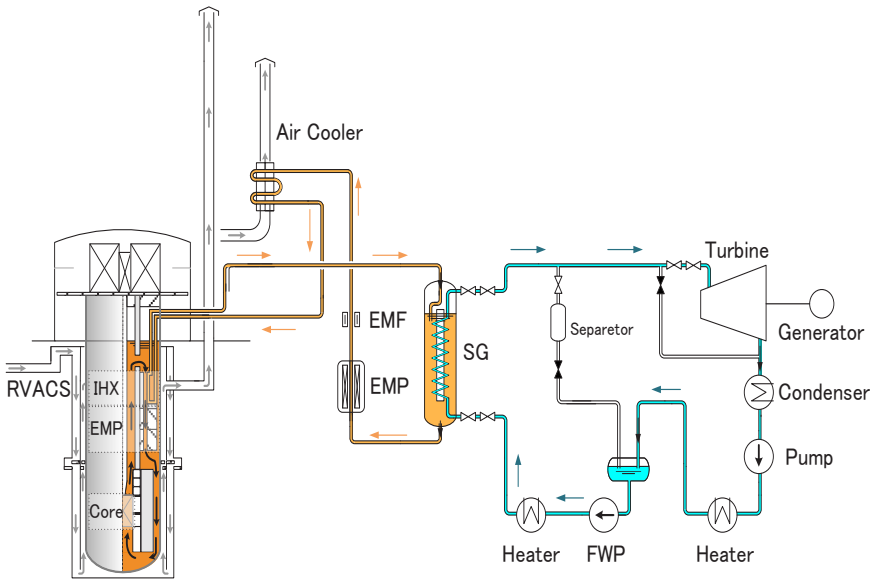


Figure 2: Simplified schematic diagram of 10 MW(e) 4S plant (Reproduced courtesy of Toshiba Corporation)

4. General Design Description

Design Philosophy

The 4S reactor is an integral pool type with all the primary components installed inside the reactor vessel (RV). Major primary components consist of intermediate heat exchangers (IHX), primary electromagnetic pumps (EM pump), moveable reflectors which form a primary reactivity control system, the ultimate shutdown rod, radial shielding assemblies, the core support plate, coolant inlet modules and fuel subassemblies. The 4S design is optimized to achieve the following design targets based on conventional sodium-cooled fast reactor technology with some innovative technologies:

- Improvement of the public acceptance and safety: all reactivity feedback are zero or negative and inherent safety features with metal fuel;
- Minimization of fuel cost and operation and maintenance (O&M) cost; ensuring enhanced proliferation resistance; long refuelling intervals (30 years for the 10 MW(e)-4S and 10 years for the 50 MW(e)-4S);
- Taking into account political circumstances: use of uranium fuel with the enrichment by U_{235} less than 20% (by weight);
- Securing fuel integrity under long-life operation of the core: adequate fuel burn-up;
- Minimization of construction costs: reduction of core size.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	Toshiba Corporation
Country of origin	Japan
Reactor type	Liquid metal cooled fast reactor (Pool type)
Electrical capacity (MW(e))	10
Thermal capacity (MW(th))	30
Expected capacity factor (%)	> 95
Design life (years)	60
Plant footprint (m^2)	10000
Coolant/moderator	Sodium
Primary circulation	Forced circulation
System pressure (MPa)	Non-pressurized
Core inlet/exit temperatures (^oC)	355 / 510
Main reactivity control mechanism	Axially movable reflectors / Fixed absorber
RPV height (m)	24
RPV diameter (m)	3.5
RPV (metric ton)	100
Configuration of reactor coolant system	Integral type (pool type)
Power conversion process	Indirect Rankine cycle
Fuel type/assembly array	Metal fuel (U-Zr alloy) based on enriched uranium
Fuel assembly active length (m)	2.5
Number of fuel assemblies	18
Fuel enrichment (%)	< 20
Fuel burnup (GWd/ton)	34
Fuel cycle (years)	N/A
Cogeneration capability	Yes
Approach to engineered safety systems	Hybrid (Active + Passive) system
Number of safety trains	3
Refuelling outage (days)	N/A
Distinguishing features	Core lifetime of ~30 years without on-site refuelling, power control by the water/steam system without affecting the core operation, passive walkaway safety.
Modules per plant	1
Target construction duration (months)	24 months for on- site construction period
Seismic design	Seismic isolator
Core damage frequency (per reactor-year)	1.7E–8
Design Status	Detailed design

Nuclear Steam Supply System

The nuclear steam supply system (NSSS) consists of the primary cooling system, the intermediate heat transport system and the water/steam system. The intermediate heat transport system has an EM pump, piping and a steam generator (SG). The SG is a helical coil type with wire-meshed double-wall tube to prevent a sodium-water reaction in the event of the tube failure.

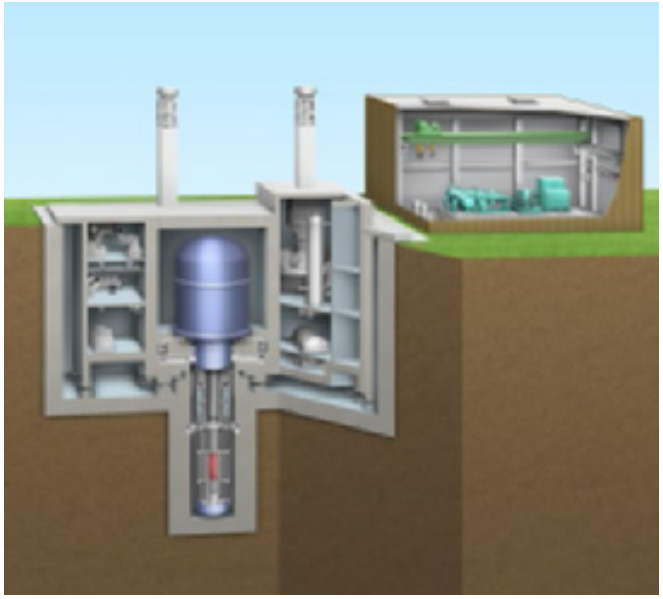


Figure 3: Under grade reactor building of the 4S plant
(Reproduced courtesy of Toshiba Corporation)

Reactor Core

The core and fuel are designed to eliminate the need for refuelling during approximately thirty (30) years for the 10 MW(e)-4S and to make all reactivity temperature coefficients negative. Metal fuel, which has an excellent thermal conductivity, is applied. The core can be operated during thirty (30) years by axially moving reflectors installed at the outside of the core, upward from the bottom. No reloading or shuffling of fuel is required during the whole core lifetime. The fuel element (fuel pin) consists of fuel slugs of U-Zr alloy, bonding sodium, cladding tube, and plugs at both ends. A gas plenum of an adequate length is located at the upper region of the fuel slugs. In the fuel subassembly, fuel pins are assembled and a top shield is installed to prevent activation of the EM pumps and the secondary sodium in the IHX. Coolant inlet modules located beneath the fuel subassembly provide a lower shielding for the reactor internal structures including the core support plate and air in the RVACS.

Reactivity Control

The reactivity control during normal operation is by the axial movement of reflectors and using fixed absorbers. The movable reflector surrounding the core gradually moves, compensating the burn-up reactivity loss over the 30 years lifetime. Therefore, the reactivity control is unnecessary at a reactor core side and this is an important factor to simplify the reactor operation. The transient overpower is prevented by the limitation of high-speed reactivity insertion by adopting the very low speed driving system.

Reactor Pressure Vessel and Internals

The RV houses all the major primary components (integral type) including the IHX, the primary EM pumps, the moveable reflectors which form a primary reactivity control system, the ultimate shutdown rod which is a back-up shutdown system, radial shielding assemblies, core support plate, coolant inlet modules and fuel subassemblies. The RV provides a primary boundary for the primary sodium coolant, and is designed with a pressure/temperature of 0.3 MPa/550°C. The design lifetime of the RV is 60 years as well as the other components.

Reactor Coolant System

The primary sodium circulates from the EM pumps downward, driven by its pump pressure, and flows through radial shielding assemblies located in the region between the RV and the

cylindrical dividing wall. The coolant flow changes its direction at the bottom of the RV and then goes upward, mainly into the fuel subassemblies and partly into the movable reflectors. The coolant flow is distributed appropriately to fuel subassemblies of each type and to the movable reflectors. Here, the core barrel separates the core and the reflector regions. Heat produced in the core is transferred to the coolant while it flows through the fuel pin bundles. The reflectors are also cooled so that the temperature becomes sufficiently low and the temperature distribution is flattened to maintain integrity through the plant life time. The coolant gathers at the hot plenum after flowing through the fuel subassemblies and the reflectors. The heated primary sodium then goes into the IHX to transfer heat to the secondary sodium. During normal operation, the primary system is enclosed inside the RV; sodium coolant is circulated by two EM pump units arranged in series. The heat generated in the reactor is transferred to the secondary sodium via the IHX located at the upper region in the RV. The secondary sodium is circulated by one EM pump unit. The heat is transferred to the water/steam system via heat transfer tubes in the SG. The heated water/steam is circulated by the feedwater pump.

Secondary System

The secondary sodium loop acts as an intermediate heat transport system and consists of the EM pump, piping, dump tank, and the SG. The secondary sodium coolant heated in the IHX flows inside the piping to the SG where heat is transferred to water/steam to be supplied to the steam turbine generator.

Steam Generator

The 4S adopts a once through type double-wall tube SG with failure detection systems. The heat transfer tube of the SG is a double-wall type. Between the inner and outer tube, wire meshes are installed, which are filled with helium, to detect one side tube failure prior to failure of the other side tube. It enables to prevent sodium-water reaction.

Pressurizer

The 4S is a sodium cooled fast reactor that does not need to pressurize inside the primary coolant boundary. Hence it has no in-vessel pressurizer.

5. Safety Features

The philosophy of the 4S safety concepts is to put an emphasis on simplicity achieved using passive and inherent safe features as a major part of the defence in depth (DiD) strategy. The ultimate objective of the 4S safety concept is to eliminate the requirement of evacuation as an emergency response measure. The 4S safety concept provides for three functions in each phase of the abnormal operation or an accident: prevention; mitigation; and confinement of radioactive material. The safety systems of the 4S consist of: redundant shutdown system; passive decay heat removal system without external power supply; emergency power system; and a reinforced reactor building. The active and passive/inherent safe features of the 4S are applied with the following main objectives:

- Reduce the probability of component failure;
- Prevent core damage in accidents;
- Two fully passive heat removal systems;
- Confine the radioactive materials;
- Prevent sodium leakage and to mitigate the associated impact if it occurs.

Engineered Safety System Approach and Configuration

In addition to the inherent safe features, there are two independent systems for reactor shutdown. The primary shutdown system provides for a drop of several sectors of the

reflector, and the back-up shutdown system provides for insertion of the ultimate shutdown rod from a fully out position at the core centre. The reflectors and the shutdown rod are fallen by gravity on scram. Either the reflector or shutdown rod is capable of enough negative reactivity to shutdown the reactor.

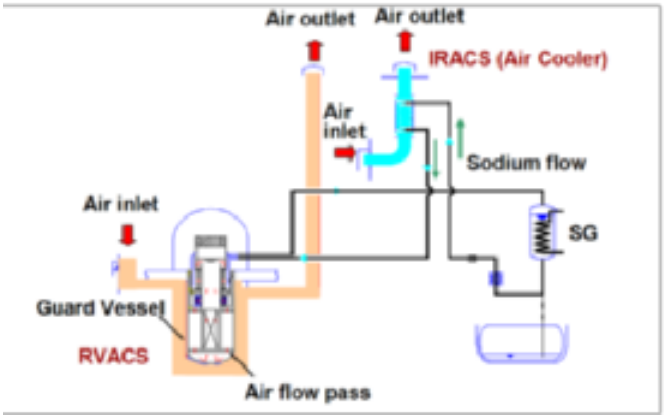


Figure 4: Engineered safety system schematic for decay heat removal of 4S (Reproduced courtesy of Toshiba Corporation)

Decay Heat Removal System

The water/steam system is available for normal shutdown heat removal. The decay heat of the core is transferred to water/steam system via the intermediate heat transport system by forced convection and is finally removed from condenser. For decay heat removal during water steam system is not available upon accidents, two independent passive systems are provided; the reactor vessel auxiliary cooling system (RVACS) and the intermediate reactor auxiliary cooling system (IRACS). The RVACS is completely passive and removes decay heat from the surfaces of the guard vessel (GV) using natural circulation of air. There is no valve, vane, or damper in the flow path of the air; therefore, the RVACS is always in operation, even when the reactor operates at rated power. Two stacks are provided to obtain a sufficient draft. The IRACS removes decay heat by air cooler which is arranged in series with the secondary sodium loop. Heat is removed by forced sodium and air circulation at the IRACS when electric power is available. In addition, the IRACS can also remove the required amount of heat solely through natural circulation of both air and sodium during loss of power events.

Emergency Core Cooling System

Pool-type sodium cooled fast reactors usually have RV and GV to keep core immersed in primary coolant even if RV failure occurs since GV keeps coolant in it. Also, primary coolant cannot be forced out from RV since there is no high pressure in RV unlike light water reactors (LWR) and all primary coolant is contained in RV. In addition, 4S has passive reactor cooling system, RVACS as mentioned above. It plays a role as an emergency core cooling system. There is no need to have LWR-like emergency core cooling system.

Containment System

The 4S adopts a cylindrical/spherical containment system. The containment system consists of the GV and the top dome, which covers the upper region of the RV, a shielding plug and the equipment located on the shielding plug. The GV provides the second boundary for the primary sodium at the outer side of the RV. For the mitigation of sodium fire, nitrogen gas is provided inside the top dome.

6. Plant Safety and Operational Performances

The 4S is designed to operate safely without active involvement of the plant operators. The design features to support such operation are:

- Burn-up reactivity swing is automatically compensated by the fine motion reflectors;
- There is no need in reloading and shuffling of fuel in the course of 30 years for the 10 MW(e)-4S;
- A reduction in maintenance requirements achieved by adopting static devices such as EM pumps or static devices continuously monitored by simple systems;
- Reduction of in-service inspections achieved by taking advantage of the non-pressurized systems of sodium-cooled reactor and by applying a continuous monitoring process based on leak-before-break detection concept to ensure safety.

7. Instrumentation and Control Systems

The instrument and control system consists of safety related and non-safety related system. The system related to safety includes the reactor protection system (RPS), the engineering safety feature actuation system (ESFAS) and the remote shutdown system (RSS). These systems have the safety class1E instruments. The RPS is plant protection system to initiate reactor trip at abnormal plant operation condition. The trip parameters are as follows:

- Reactor core neutron flux;
- Liquid sodium level in the RV;
- Supply voltage/current of main/intermediate loop EM pumps;
- Primary outlet temperature of IHX;
- Voltage of power line;
- Seismic acceleration.

The RSS provides means to shut down the reactor from outside the main control room. The system related to non-safety consists of the plant control system (PCS) and the interface reactor system (IRS).

8. Plant Arrangement

The plant layout of the 4S is optimized to meet various functional needs; the requirements for safety, radiation zoning, piping and cabling; construction requirements; access and security considerations. The general philosophy of the 4S plant layout is as follows:

- Efficient space utilization and minimization of volume of the buildings;
- Horizontal seismic isolation for the reactor building;
- An embedded reactor building, securing that the reactor is earth-sheltered;
- Lightweight buildings to assure a high degree of transportability in construction;
- The secondary sodium loop area is categorized as a non-radiation-controlled area; to achieve this, a sufficient shielding for the IHX is provided.

Reactor Building

The 4S is a land-based nuclear power station with the reactor building embedded underground for security reasons, to minimize unauthorized access and enhance inherent protection against extreme external events. The reactor building is supported by horizontal seismic isolators, reinforced and protected from massive water invasion by keeping its water-tightness. The reactor building including the concrete silo can be used for more than 60 years.

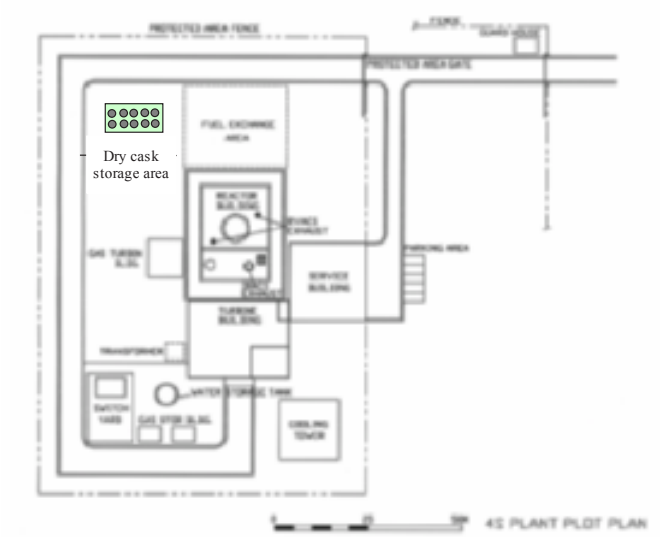


Figure 5: The 4S plant layout diagram (Reproduced courtesy of Toshiba Corporation)

Control Building
The control room is settled in the reactor building which is supported by the horizontal seismic isolators and embedded underground.

Balance of Plant:
The BOP including a steam turbine system is located at ground level.

Turbine Generator Building
The 4S plant consists of one reactor and one turbine generator system. Superheated steam is supplied from the steam generator to the turbine.

Electric Power Systems
These systems include the plant main generator (PMG), the main power transformer and the generator circuit breaker (GCB), diesel power generator and batteries. The grid is also connected to the unit auxiliary transformer (UAT). The PMG supplies the power to the onsite power subsystem via the UAT. The two class 1E buses are separated from each other and separated from the Non-class 1E electric system. Each class 1E system is provided with a separate emergency diesel generator and batteries.

9. Design and Licensing Status

Licensing activities for the 4S design initiated with the U.S.NRC in 2007. In pre-application review, four (4) meetings had been held in the past and fourteen (14) technical reports have been submitted to NRC. Toshiba is conducting the detailed design and safety analysis for design approval. In parallel, Toshiba continues to look for customers.

10. Plant Economics

To be a viable option for power generation in remote areas, the 4S should provide a competitive cost of electric power. The target plant construction cost is US$ 10000/kWe and the O&M cost of 4.5¢/kWh. The target construction cost could be achieved by the simplified and standardized design, shop fabrication, modular construction, short construction time, and mass production. The O&M cost could be achieved by long-refuelling interval, and low-maintenance requirements of components such as EM pump.

BREST-OD-300 (NIKIET, Russian Federation)

1. Introduction

BREST-OD-300 is a lead cooled fast reactor fuelled with uranium plutonium mononitride (PuN–UN) that uses a two circuit heat transport system to deliver heat to a supercritical steam turbine and generate electricity of 300 MW for experimental and demonstration purposes. BREST-OD-300 is the reference Russian design of a medium-size lead cooled fast reactor. The lead-cooled fast reactor is one of the alternative fast reactors under development in the country. Russian Federation has the operational experience on the use of lead-bismuth eutectic alloy in power reactors for submarine propulsion, using the SVBR-80 technology. This experience is being incorporated in the development of lead cooled fast reactors. The present goal of the project is to implement all necessary R&D in order to finalize the detailed design of the BREST-OD-300.

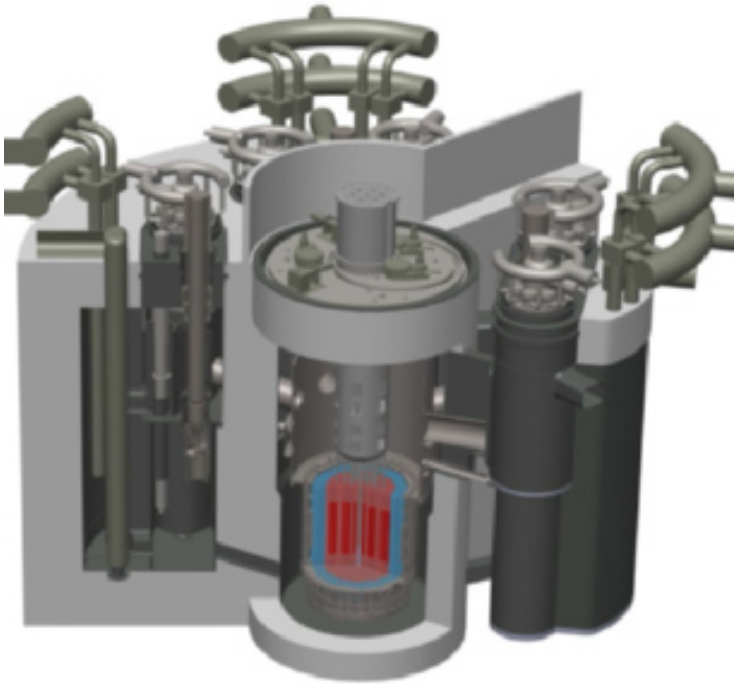


Figure 1: Reactor system configuration of BREST-OD-300
(Reproduced courtesy of NIKIET)

2. Target Applications

The BREST-OD-300 power unit is designed as a pilot and demonstration unit intended for studying the reactor facility operation in different modes and optimizing all processes and systems that support the reactor operation. After operational tests, the unit will be commissioned for electricity supply to the grid. The multi-purpose application configuration of BREST is shown in *Figure 2*.

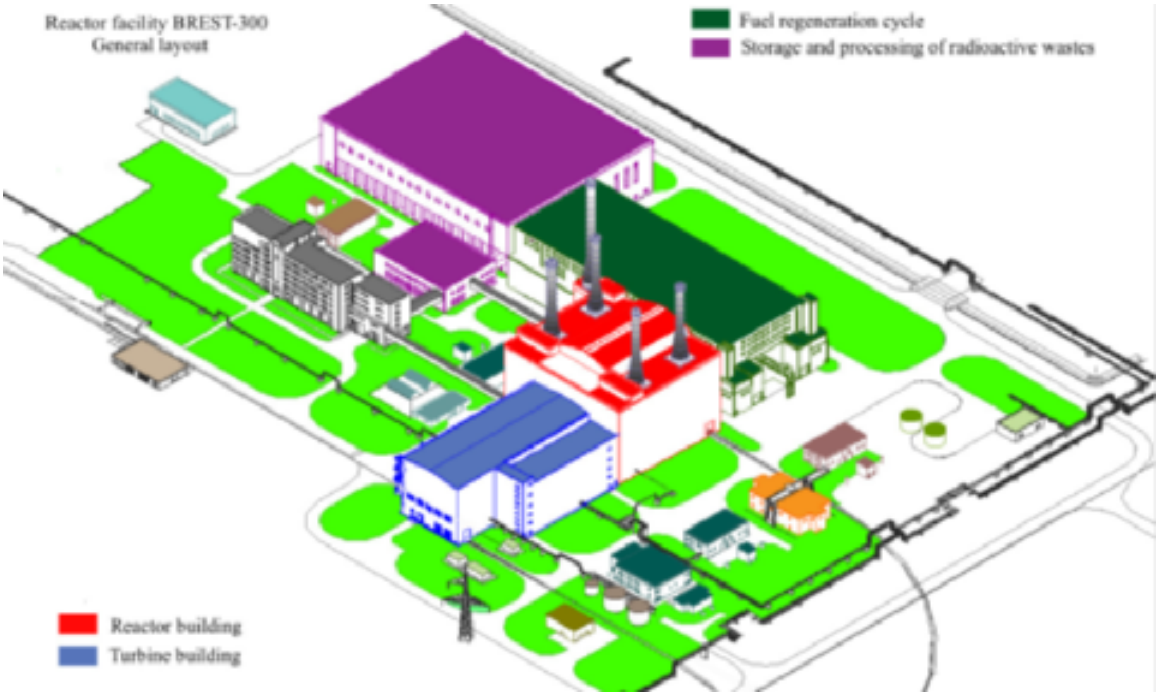


Figure 2: BREST multi-purpose application configuration (Reproduced courtesy of NIKIET)

3. Development Milestones

1995	Conceptual design development initiated
2002	A feasibility study of the BREST-OD-300 NPP with an On-Site Nuclear Fuel Cycle (OSNFC) for the Beloyarsk NPP Site
2020s	First of a kind engineering demonstration plant

4. General Design Description

Design Philosophy

BREST-OD-300 is a pool type reactor design not requiring metal vessel, hence, not limiting reactor power. The objective of the design is to eliminate severe accidents; complete fuel breeding (equilibrium mode) for self-sustaining; integral-type arrangement of the first circuit to avoid output of coolant outside the reactor vessel, to eliminate loss of coolant; use of low-activated coolant with high enough boiling temperature, without rough interaction with water and air in the case of depressurizing of the circuit. The reactor facility has a two-circuit steam generating power unit that includes a reactor, steam generators (SG), main coolant pumps (MCP), fuel assembly (FA) loading system, control and protection system (CPS), concrete vault with a heat shield, steam-turbine unit, cool-down heat removal system, reactor heat-up system, reactor overpressure protection system, gas purification system and other auxiliary systems.

Reactor Coolant System

BREST-OD-300 uses a mixed integral primary circuit configuration as the SGs and MCPs are installed outside the reactor central vessel. As compared to the integral traditional fast reactor designs, the BREST concept has reduced dimensions and volume of the primary circuit. The reactor and the SGs are located in the thermally shielded concrete vault, without using a metal vessel. The concrete temperature is kept below the allowable limit by means of natural air circulation. Heat is removed from the reactor core through forced lead coolant (LC) circulation by pumps. The LC is pumped to the height of ~2 m relative to the lead level

in the suction chamber and supplied to the free level of the annular pressure chamber. The lead further goes down to the core support grid, flows upward through the core where it is heated up to the temperature of 540°C, and is supplied to the common "hot" coolant drain chamber. Then it goes up and flows over the SG inlet cavities and inter-tube space via the distribution header nozzles. As it goes down in the inter-tube space, the LC transfers heat to the secondary coolant flowing inside the SG tubes. Cooled-down to ~ 420°C, the LC goes up in the annulus and flows out the pump suction chamber, where from it is pumped again to the pressure chamber shown in *Figure 3*.

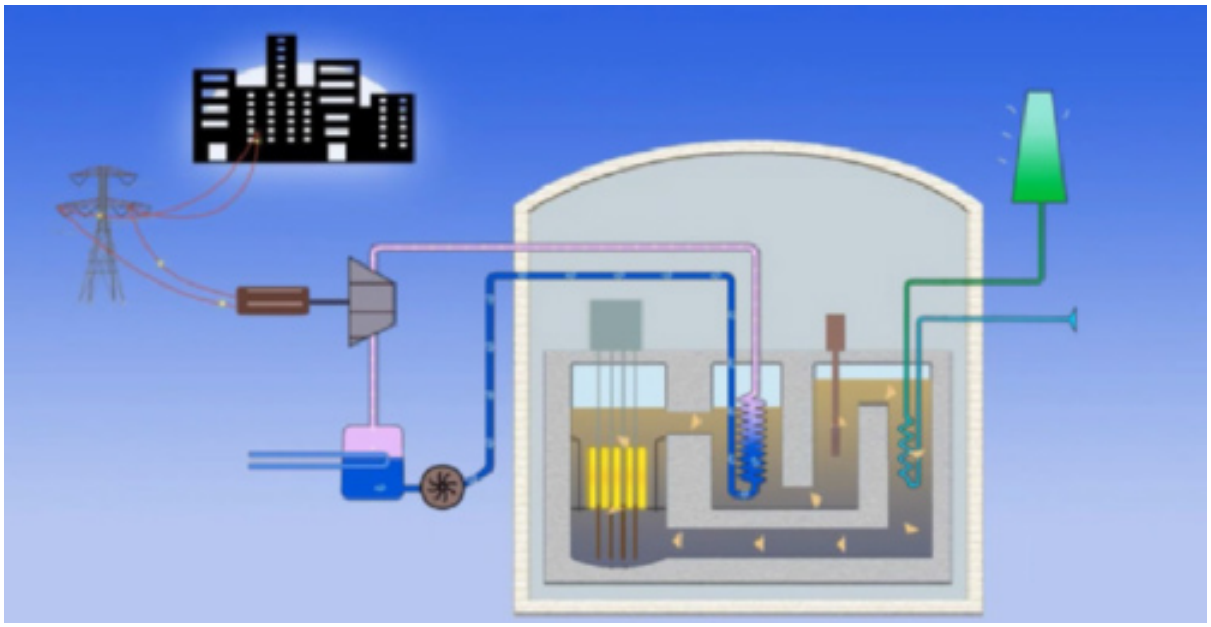


Figure 3: Schematic Flow Diagram of BREST-OD-300 (Reproduced courtesy of NIKIET)

The absence of high pressure in the primary lead circuit and a relatively high lead freezing temperature contribute to crack self-healing, which excludes loss-of-core-cooling accidents and outflow of radioactive lead from the reactor vessel. Lead circulation through reactor core and steam generator takes place due to the difference between the levels of cold and hot coolant developed by the pumps, rather than the head developed by the pumps. Non-uniformity of lead flow through the steam generators with one of few pumps shut down is excluded, in so doing flow inertia in fast pump shutdown is provided by equalizing coolant levels in discharge and suction chambers. As for primary coolant flow system, while flowing the coolant reaches free level twice, that results in surfacing and escaping majority of lead vapour bubbles, being generated in accidents caused by steam generator tube depressurization.

Nuclear Steam Supply System

The primary cooling system is based on forced circulation during normal operation. The system has natural circulation capability for emergency condition. The integral BREST design of high-boiling point (~2000 K) and low-activated lead coolant, does not require high pressure in the primary circuit, and excludes the potential loss of coolant accidents.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology Developer	NIKIET
Country of origin	Russian Federation
Reactor type	Liquid metal cooled fast reactor
Electrical capacity (MW(e))	300
Thermal capacity (MW(th))	700
Steam production rate (t/hr)	1480
Expected Availability Factor (%)	82
Design Life (years)	30
Plant Footprint (m^2)	65 x 74
Coolant/moderator	Lead
Primary circulation	Forced circulation
Number of reactor coolant pump	4
System pressure (MPa)	Low pressure operation
Core inlet/exit temperatures (^oC)	420/540
Main Reactivity Control Mechanism	Shim and automatic control rods ($\Delta\rho \approx 14\ \beta_{eff}$)
RPV Height (m)	14.14
RPV Diameter (m)	6.88
Module Weight (metric ton)	8600
Configuration of Reactor Coolant System	Integrated
Power Conversion Process	Indirect Rankine Cycle
Fuel Type/Assembly Array	–
Fuel material	PuN-UN
Fuel Assembly Active Length (m)	1.1
Number of Fuel Assemblies	145
Fuel Enrichment (%)	~ 13.5
Fuel Burnup (GWd/ton)	61.45
Fuel Cycle (months)	60-72, with partial refuelling – 1/year
Breeding ratio	1.05
Cogeneration Capability	Yes
Approach to Engineered Safety Systems	• Passive and time-unlimited direct residual heat removal from the lead circuit system via natural air circulation with heat discharge into the atmosphere • Emergency protection rods ($\Delta\rho \approx 6\ \beta_{eff}$)
Number of safety trains	–
Refuelling Outage (days)	–
Distinguishing features	High level of inherent safety due to natural properties of the lead, fuel, core and cooling design
Modules per plant	1
Target Construction Duration (months)	–
Seismic design	VII-MSK 64
Predicted core damage frequency (per reactor year)	–
Design Status	Detailed design; for potential start-up in early 2020s

Reactor Core

The reactor of a pool-type design has an integral lead circuit accommodated in one central and 4 peripheral cavities of the concrete steel-lined vessel. The central cavity houses the core barrel together with the side reflector, the CPS rods, the spent fuel assembly (SFA) storage and the shell that partitions the hot and the cold lead flows. Four peripheral cavities (according to the loop number) accommodate the SG-MCP units, heat exchangers of the emergency and normal cool-down systems, filters and auxiliary components. The cavities have hydraulic interconnection. The BREST reactor core has a high-boiling point of ~1725°C, radiation-resistant, low-activated lead coolant, which is chemically inert with water and air, does not require high pressure in the primary circuit, and excludes the potentiality of accidents with a loss of coolant and heat removal, fires and explosions in a contact with the environment. The lead coolant properties in combination with a dense, heat-conducting nitride fuel provide conditions for complete plutonium breeding in the core (CBR ≥ 1). That results in a small operating reactivity margin ($\Delta\rho < \beta eff$) and enables power operation without prompt-neutron reactor power excursions.

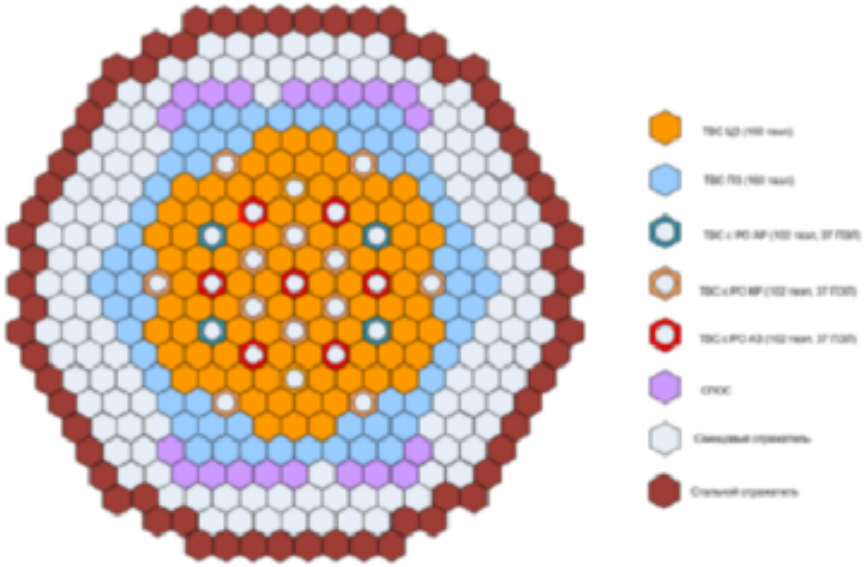


Figure 4: The BREST-OD-300 reactor core (Reproduced courtesy of NIKIET)

The adopted fuel is mixed mononitride (UN-PuN) which exhibits high density (14.3 g/cm^3) and high conductivity (20 W/m·K) and is compatible with lead and the fuel cladding of chromium ferritic-martensitic steel. In order to reduce the fuel temperature and thus make the release of fission products from the fuel relatively low, the gap between the fuel and the cladding is filled with lead which ensures a suitable thermal fuel-coolant contact. To provide a significant coolant flow area, increase the level of power removed by natural lead circulation, reduce the coolant preheating temperature and, primarily, exclude the cooling losses in the damaged FA in case of local flow rate blockage, all core FAs do not have shrouds. The FA design allows radial coolant overflow in the core which prevents overheating of the damaged FA. The FA design allows radial coolant overflow in the core that prevents overheating of the damaged FA. Better radial uniformity of FA power levels and coolant heating is ensured by regionalized fuelling and lead flow rates by placing assemblies with smaller-diameter rods in central regions, and larger-diameter in peripheral regions. The adopted fuel exhibits high density and high heat conductivity, and is compatible with lead and the fuel cladding of chromium ferritic-martensitic steel. The reflector assemblies have leak-proof cans. The first row of these assemblies is used as control element channels and the assemblies in rows 2 to 4 can contain long-lived iodine and technetium for transmutation and

also contain Sr_{90} and Cs_{137} as a stable heat source for prevention of hypothetical coolant freezing. Instead of traditional flattening of the radial power density distribution by means of different fuel enrichments, this is achieved by means of a core composed of three zones with fuel elements having different diameters, but the same plutonium content. Such a method guarantees a suitable flattening of the lead temperatures at the core outlet, as well as the respect of the maximum fuel cladding temperatures. To reduce the neutron leakage, to better levelling the neutron flux, ensure the operation conditions without cycle reactivity change and attain the complete fuel breeding in the core (CBR ~ 1), traditional uranium shields have been replaced with an effective lead reflector whose albedo characteristics are better than those of uranium dioxide.

Reactivity Control

Reactivity control during normal operation is achieved using shim and automatic control rods ($\Delta\rho \approx 14\ \beta_{eff}$). Emergency protection rods are also provided ($\Delta\rho \approx 6\ \beta_{eff}$).

Secondary Circuit

The use of chemically inert, high-boiling molten lead in the primary circuit allows adoption of a two-circuit unit configuration, with a supercritical steam system as secondary circuit. The secondary circuit is a non-radioactive circuit consisting of SGs, main steam lines, a feedwater system and one turbine unit with supercritical steam parameters. A standard K-300-240-3 turbine unit with three-cylinder (HPC+MPC+LPC) steam condensation turbine with intermediate steam superheating and a rotation speed of 3000 rev/min is used. The nominal steam flowrate to the turbine is about 1000 t/h. Oxygen neutral water at supercritical pressure is used in the secondary loop. Such provision does not require removal of oxygen and make the deaerator in the secondary circuit unnecessary. The intermediate steam superheating system contains two steam-to-steam heat exchangers (SSHX). The SSHX heating fluid is steam extracted from the main steam lines upstream of the turbine at a flowrate of 600 t/h. Downstream of the SSHX, all cooled fluid flowrate is supplied via throttling control valves as the heating fluid to the feedwater mixing preheater (FWMP) where it additionally heats the water downstream of the high pressure preheater to the temperature of 355°C at the pressure of 170 kg/cm^2. The feedwater preheating at such a temperature is necessary to prevent the lead coolant freezing (T_{melt} = 327°C) in the primary circuit in emergency conditions. It is for the same purpose that cutoff valves (two for each SG) are installed on the pipelines for feedwater supply to the SGs. They are closed automatically both by the "passive drive" and the "active drive" in response to respective emergency signals. Feedwater pumps (FWP-2) (two working pumps and one standby pump) supply feedwater from the FWMP to each SG via control units with the main and starting control valves. The FWP-2 speed is controlled in the range from 70 to 100% of the nominal value. Each FWP-2 is equipped with a recirculation line. The common header serves as a starting recirculation line to the FWMP with a capacity of 350 t/h and a control valve for warming up the entire system prior to the water supply to the SG. The peculiarity of the secondary circuit is that it is not charged with safety functions for emergency heat removal from the reactor.

5. Safety Features

The principle of natural safety is achieved through the inherent physical and chemical behaviour and properties of fuel, coolant and other reactor components, negative reactivity feedbacks, natural dependencies and neutron balance in fast reactors. Accidents are avoided due to the intrinsic safety features of BREST, including reactivity fuel temperature coefficient, coolant and core design components, and also coolant pressure and temperature at the core inlet and outlet. For this reason BREST can be considered an intrinsically safe reactor. However, the search for the weaknesses in the design of the reactor and its safety

systems in terms of safety continues, and it is not excluded that the list of predominantly passive engineered features for overcoming accidents caused by previously ignored initial events will be expanded. An accident with a SG tube rupture is one of the most adverse events for BREST-OD-300. To reduce the consequences of a potential accident with a SG tube rupture, a mixed integral/loop configuration of the primary circuit is adopted, with SGs and MCP installed outside the reactor central vessel. Together with the selected lead circulation pattern and steam dump from the reactor gas volume to the localization system, such configuration excludes the ingress of the hazardous steam into the core and reactor overpressure.

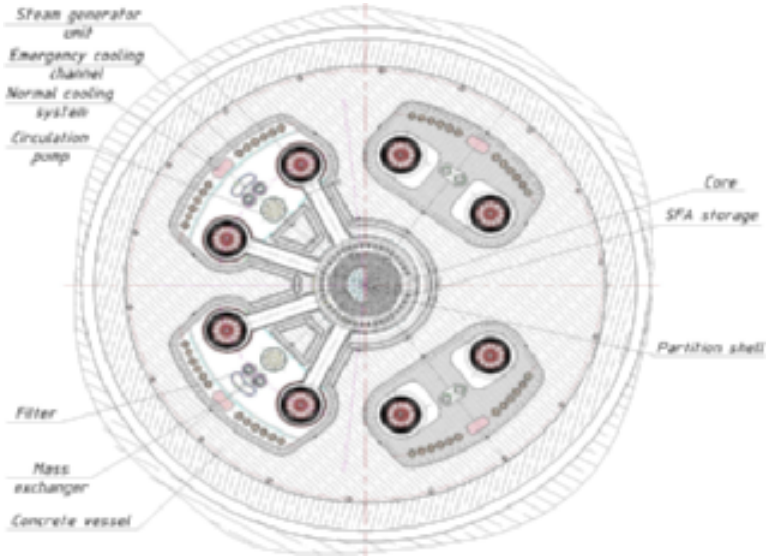


Figure 5: Cross sectional view of the BREST (Reproduced courtesy of NIKIET)

Emergency Core Cooling System

The emergency core cooling system (ECCS) uses field pipes, immersed directly in lead, which may be used to cool down reactor under normal conditions. The system coolant circulation in emergency heat removal mode is by natural circulation, with the system coolant under atmospheric pressure. The system consists of 4 loops. The ECCS air circuit inlet air temperature operates at a minimum and maximum temperature of –55°C and 37°C respectively. By-pass line incorporated in the coolant loop prevents inadvertent valve actuation in pumps shut down. The safety analysis has shown that all considered initial events involving a fast introduction of reactivity up to its full margin (spontaneous movement of all CPS control rods, steam throw to the core at a SG tube rupture, etc.), interruption of the forced coolant circulation (trip or arrest of all pumps), loss of secondary heat sink or lead super cooling at the core inlet and so on, do not lead to accidents with fuel damage and inadmissible radioactive or toxic releases, even in the case of a failure of the reactor active safety systems.

6. Plant Arrangement

Plant main building consists of the reactor containment building, auxiliary building, compound building (CPB), emergency diesel generator building and turbine-generator building (TGB). The reactor building and other structures are mounted on a single monolithic reinforced concrete foundation plate. In order to reduce seismic inertia forces, the building is designed to be symmetrical with the footprint of 65×74 m and separated with intercoupled vertical bearing diaphragms shown in *Figure 2*. The main building's structures have been designed based on the "Regulations for Aseismic Design of Nuclear Power Plants". The

magnitude of 6 is assumed in the design as the design basis earthquake and the magnitude of 7 on the MSK-64 scale is assumed as the ultimate design-basis earthquake. The buildings are separated from each other with an aseismic temperature-deformation joint.

Reactor Building
The reactor building and fuel storage area are equipped with a full monitoring system with closed circuit monitoring system (CCTV).

Balance of Plant:

Turbine Generator Building
The reference concept of the turbine plant has been developed.

Electric Power Systems
These systems include the main generator, main transformer, unit auxiliary transformers, stand-by auxiliary transformers, diesel generators, and batteries. The electrical systems, including the class 1E and non-class 1E, are based on a two train approach.

7. Design and Licensing status

The development and construction of the BREST-OD-300 reactor is included in the framework of tasks in:

- Development Strategy of Nuclear Power in the Russian Federation in the first half of the 21st century approved by the Russian Government in 2000;
- The Federal Target program "Nuclear Power Technologies of a new generation for the period of 2010-2015 and up to the year 2020" approved by the Russian Government in 2010;
- The "Proryv" project (2011) that integrates projects on the strategic solution of target tasks on the creation of natural-safety nuclear power technologies based on fast-neutron reactors and a closed nuclear fuel cycle (CNFC).

The BREST-OD-300 power unit is designed as a pilot and demonstration unit intended for studying the reactor facility operation in different modes and optimizing all processes and systems that support reactor operation. Furthermore, BREST-OD-300 is considered the prototype of a fleet of medium sized power reactors.

8. Plant Economics

A plant with a BREST-type reactor is expected to be economically competitive owing to the simpler design of the facility and its safety systems, as well as to efficient utilisation of nuclear fuel and generated heat. The integral configuration of the circuit components in a concrete pool predicts a reduced construction costs. On-site fuel cycle arrangement is also likely to be economically beneficial owing to the shorter out-of-pile cooling and transportation time, which will eventually lead to a reduction in the recycled fuel quantity – one of the greater contributors to the fuel cycle costs. Comparative economic estimates allow one to hope to the expenses not higher than

SVBR-100 (JSC AKME Engineering, Russian Federation)

1. Introduction

The SVBR-100 is a multipurpose small modular fast reactor lead–bismuth (LBE) cooled with an equivalent electric power of 100 MW. In the Russian Federation, lead–bismuth cooled reactor technology has been used in several nuclear submarines. The SVBR technology, according to its basic parameters and salient technical characteristics, is claimed as a Generation IV nuclear reactor. The development of SVBR-100 is based on the experience gathered in the design and operation of several LBE facilities on nuclear submarines (NSs), which allows:

- Use of mastered LBE technology;
- Use of almost all basic components, units and equipment devices of the reactor installation, which are verified by operational experience in LBE;
- Capability to master primary and secondary circuits;
- Use of existing fuel infrastructure;
- Ensuring the corrosion resistance of structural materials;
- Controlling the LBE quality and the mass transfer processes in the reactor circuit;
- Ensuring the radiation safety of personnel carrying out work with equipment contaminated with the Po_{210} radionuclide; and
- Multiple LBE freezing and unfreezing in the reactor facility.

2. Target Application

The possibility of multi-purpose application of modular nuclear power plants (NPP) of different capacities (100 – 600 MW) based on SVBR-100 creates the conditions for satisfying the requirements of consumers in a new sector of regional and small-scale atomic energy industry: creation of regional NPP and nuclear co-generation plant (NCGP) of low and medium capacity utilization as part of floating NPPs renovation of NPP units. The standard reactor modules of 100 MW(e) can be used for multipurpose, e.g.:

- Modular NPP of small, medium or large power;
- Regional nuclear heating and electricity generating plant of 200-600 MW(e) which are located not far from the cities;
- Refurbish of NPP units whose reactors have expired their lifetime;
- Nuclear desalination systems.

3. Development millstones

2010-2016	Research & development and design of the reactor unit and the pilot plant
2016-2019	Pilot plant and supply of equipment
2019	Pilot plant physical and power generation start up
2024-2025	Serial production and supply of packaged equipment

4. General Design Description

The main components and sub-systems of the primary circuit which includes: reactor monoblock, secondary circuit, passive hear removal system (PHRS) are shown in *Figure 1*. All the equipment of the primary circuit are installed inside the vessel of the reactor

monoblock (RMB). The removable unit composed of the core, the control rods and a shielding plug is installed in the centre of the RMB and it is surrounded by an in-vessel radiation shielding (boron carbide) with SG and MCP modules placed inside it as shown in *Figure 2*.

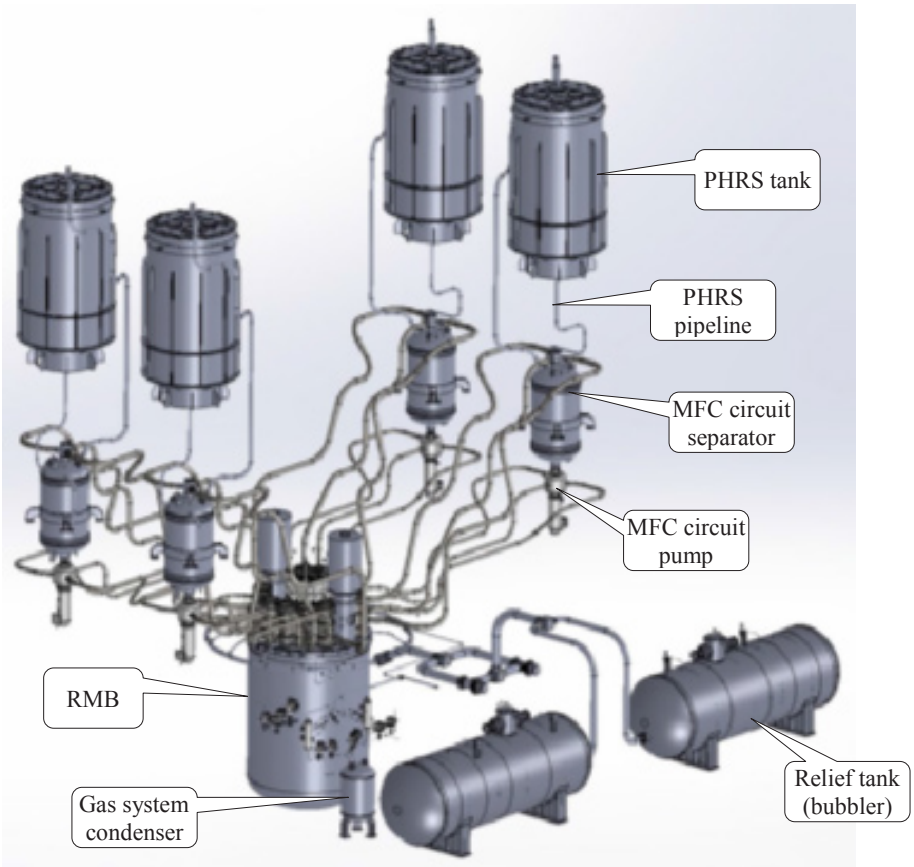


Figure 1: The SVBR-100 reactor system configuration (Reproduced courtesy of JSC AKME Engineering)

Design Philosophy

SVBR-100 is designed as a multipurpose modular integral lead-bismuth cooled small power fast reactor to generate an equivalent electricity of 100 MW(e). The design is based on more than 80 reactor-years operational experience of LBE cooled reactors for submarine propulsion application. Its main features include:

- Enhanced inherent self-protection and passive safety and significant simplification of the design of the reactor as well as entire NPP;
- Possibility to operate with different type of fuel in different fuel cycles (period of operation without refuelling: not less than 7-8 years);
- Compact design and maximum factory readiness of the reactor;
- Possibility of creation of module based structured NPP with power multiplying by adding the reactors.

Reactor Coolant System

The entire primary equipment circuit of SVBR-100 is contained within a robust single reactor vessel. A protective casing surrounds the single-unit reactor vessel. The reactor passes heat to a two-circuit heat-removal system and SG with multiple-circulation secondary coolant

system. Natural circulation of coolant in the reactor heat removal circuits is sufficient to passively cool down the reactor and prevent hazardous superheating of the core. The coolant technology system includes mass-exchangers, gas mixture ejectors, sensors of oxygen activity in LBE; its function is to maintain the LBE quality, inhibiting structural materials corrosion. The circuits for primary coolant circulation (the main and the auxiliary one), are wholly realized by components of the in-vessel components, without using pipelines and valves. Within the main circulation circuit (MCC), the coolant flows according to the following scheme, as shown in *Figure 3* (flow diagram). Being heated in the core (cross section of the core is shown in *Figure 4*), coolant flows to the inlet of the medium part of the inter-tube chamber of 12 SG modules connected in parallel to each other.

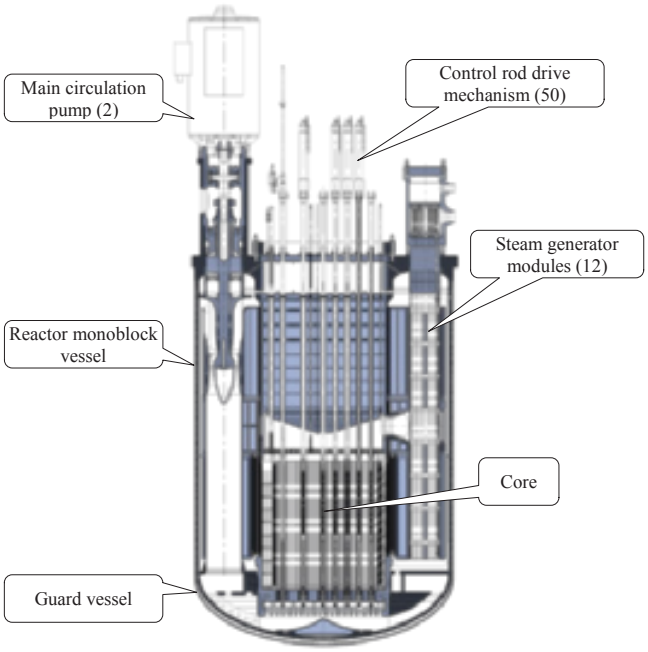


Figure 2: Equipment arrangement in the RMB vessel of SVBR-100
(Reproduced courtesy of JSC AKME Engineering)

Then coolant is divided into two flows. One flow moves upwards in the inter-tube chamber and enters into the peripheral buffer chamber with a free level of the "cold" coolant. Another flow moves downwards and enters into the outlet chamber out of which it goes to the channels into in-vessel radiation shielding. Coolant flow moves upwards through in-vessel radiation shielding and cools it, and then it enters into the peripheral buffer chamber as well. Out of the peripheral buffer chamber, the coolant flow goes over the downcomer circular channel along the RMB vessel via the inlet chamber to the MCP suction. Out of the MCP the coolant flows over the two channels installed in the mono-block of the lower zone of in-vessel radiation shielding into the distributing chamber, from which main part of flow goes to the reactor inlet chamber, thus closing the MCC circuit. Very small part of flow moves upwards via the gap near RMB vessel wall, cooling it and goes into the peripheral buffer chamber.

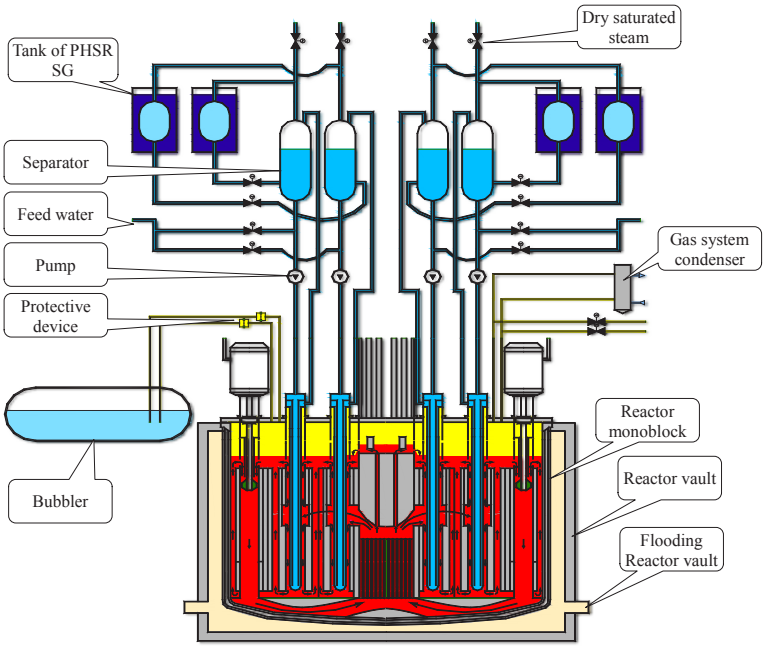


Figure 3: SVBER-100 Flow diagram (Reproduced courtesy of JSC AKME Engineering)

Secondary System

The secondary system includes: SG modules, feedwater and steam pipelines, separators and autonomous cooling condensers. The basic equipment of SVBR-100 is installed in an 11.5 m high tight box-containment. In the lower part of each box there is a concrete well to flooded water in an event of beyond the design accident the failures all four PHRS via SGs. The reactor monoblock is installed inside concrete well and is fastened on the head ring of roof. In the upper part of the box there is the reactor equipment, including four steam separators and four cooling condensers immersed in the water tanks PHRS. The high elevation of the separators has been selected in order to guarantee the coolant natural circulation in the secondary circuit in all reactor operation modes. The gas system condensers are installed in the upper part of the box in the separate concrete compartment. Arrangement RMB in the reactor vault is shown in *Figure 5*.

Reactor Core

The SVBR-100 reactor core operates without any partial refuelling. The fresh fuel is loaded as a single cartridge while the spent nuclear fuel is unloaded cassette by cassette. The core configuration allows for a lower power density compared with the nuclear submarines using LBE reactors. This design has the capability to utilize various fuel cycles. The first stage will be the typical uranium oxide fuel leading to a core breeding ratio (CBR) of 0.84; MOX fuel can also be used, leading to a CBR just about 1. Using UO_2 as the starting fuel, the closed fuel cycle can be realized in 15 years. Nitride uranium and uranium plutonium fuel can also be used to improve safety and fuel cycle characteristics. The SVBR-100 reactor pursues resistance to nuclear fissile material proliferation by using uranium with enrichment below 20% while using uranium oxide fuel in the initial core. The reactor is designed to operate for eight years without core refuelling.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	JSC Institute for Physics and Power Engineering and JSC EDB
Country of origin	Russian Federation
Reactor type	Liquid metal cooled fast reactor
Electrical capacity (MW(e))	100
Thermal capacity (MW(th))	280
Expected capacity factor (%)	< 90
Design life (years)	60
Coolant/moderator	Lead-Bismuth eutectic alloy
Primary circulation	Forced circulation
System pressure (MPa)	Low pressure
Core inlet/exit temperatures (^{o}C)	340 / 485
Main reactivity control mechanism	Control rod drive mechanism
RPV height (m)	7.55
RPV diameter (m)	4.53
RPV or module weight (metric ton)	280
Configuration of reactor coolant system	Integral type
Power conversion process	Indirect Rankine cycle
Fuel type/assembly array	UO_2
Fuel assembly active length (m)	2.1
Number of fuel assemblies	61
Fuel enrichment (%)	< 19.3
Fuel burnup (GWd/ton)	60 (average)
Fuel cycle (years)	7 – 8
Cogeneration capability	Yes
Approach to engineered safety systems	Passive
Number of safety trains	4
Refuelling outage (days)	60 days in 7-8 years
Distinguishing features	Integral (monoblock) layout of the primary circuit where all components (reactor, steam generation modules, main circulating pumps) are installed arranged in one vessel
Modules per plant	1 – 6
Target construction duration (months)	36
Seismic design (g)	0.5
Core damage frequency (per reactor-year)	1E–8
Design Status	Detailed design for potential construction in early 2020s

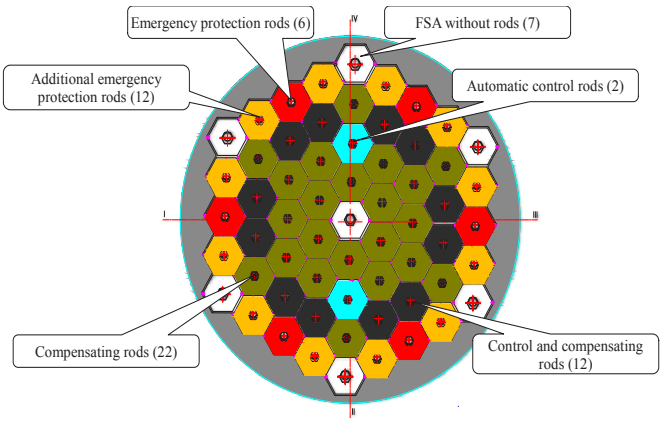


Figure 4: Cross section of the SVBR-100 core (Reproduced courtesy of JSC AKME Engineering)

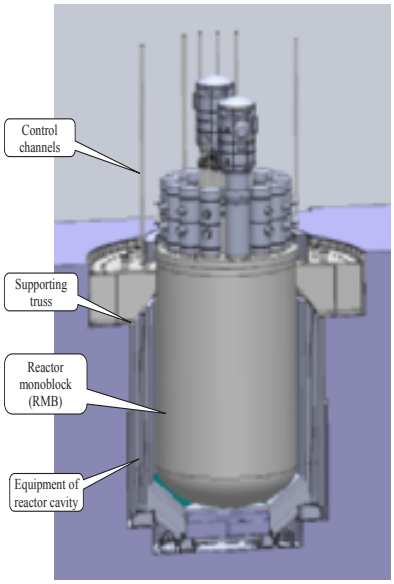


Figure 5: Reactor monoblock in the reactor vault (Reproduced courtesy of JSC AKME Engineering)

5. Safety Features

Physical basement for high level of inherent self-protection and passive safety:

- First, this is potential energy contained in coolant. LBE does not contain potential energy, which in an event of accident occurrence can cause destruction of defense barriers, core damage and disastrous exhaust of radioactivity. This is potential energy of coolant compression and potential chemical energy of coolant's interaction with sodium coolant structural materials (zirconium), water and air;
- Potential energy is a natural property of coolant and cannot be changed by any technical solutions;
- Further, it is an integral structure of the reactor facility that completely eliminates pipelines and valves with radioactive coolant and eliminates an opportunity of coolant leak;
- Finally, this is a fast neutron reactor, in which there are no poisoning effects, low burnup reactivity margin, low value of negative temperature reactivity effect, negative void reactivity effect. Efficiency of the strongest absorbing rod does not exceed 1 $, that been coupled with technical performance of the control and protection system exclude an opportunity of prompt neutrons runaway in the reactor.

SVBR-100 safety does not depend on the state of the systems and equipment of the turbine-generator installation. Distinguishing features of the SVBR-100 – Safety concept is shown in *Figure 6.*

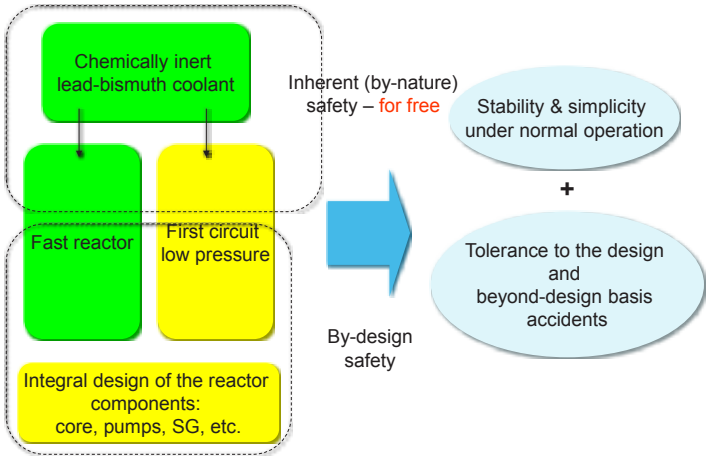


Figure 6: Distinguishing Features of the SVBR-100 – Safety Concept (Reproduced courtesy of JSC AKME Engineering)

6. Instrumentation and Control systems

The main principles of the design include:

- Distributed I&C system with several level of hierarchy and defense in depth;
- Soft control of NPP technological systems;
- Availability of the large screen and reserve zone at main control room (MCR);
- Principles of diversity, reliability, physical separation and others, providing high level of functional reliability, including protection against common cause failures;
- Well-developed diagnostic functions;
- Self-diagnostic of I&C programmable devices.

Design specificities are:

- Regulation of coolant flow rate by changing rotation speed of MCP depending on reactor power for maintaining constant coolant heat up;
- Full scope diagnostic system of NPP;
- Providing I&C operability during 7-8 years of continuous NPP operation;
- New tasks of neutron flux monitoring while core refuelling;
- Highly reliable reactor control at start up and operation;
- Load-follow operation in the deep range (100–50–100%).

7. Plant Arrangement

With the purpose of using SVBR-100 modules in large-scale NPPs, a Russian consortium of IPPE, Gidropress, FSUE Atomenergoproekt and others have developed a conceptual design of a NPP which consists of two power units of 1600 MW(e) each. A NSSS consisting of 16 SVBR-100 reactor modules and a turbine of 1600 MW(e) in each power unit is arranged in the main building. A modular design of the NSSS power unit allows have load factor (LF) not less than 90% under long reactor operation. When a reactor module is shut down for refuelling, power of the unit reduces slightly. Modular NPPs also allow fabrication of the single modules in a factory, easy water or ground transportation, include rail way, of the modules to the NPP site and reduced construction times. General view of pilot plant is shown in *Figure 7*.

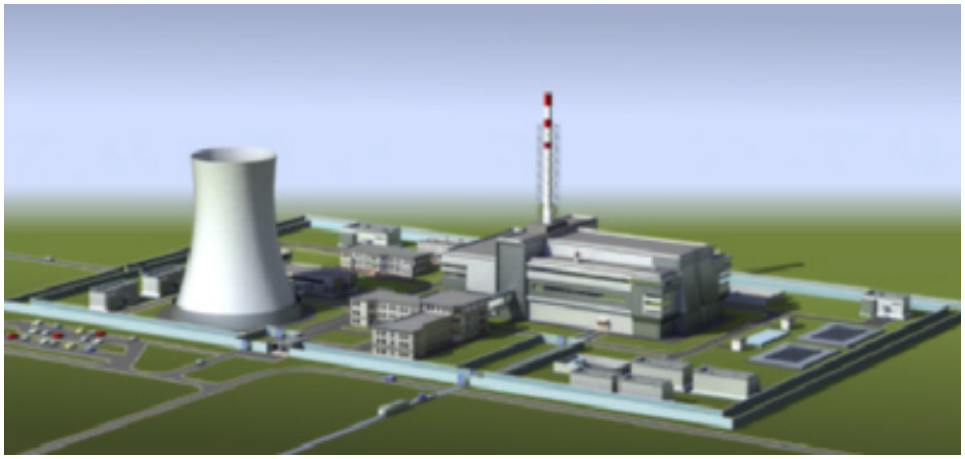


Figure 7: Pilot plant of the NPP with the SVBR-100 reactor module (Reproduced courtesy of JSC AKME Engineering)

Control building
Control building will be developed later.

Balance of plant:

Turbine Generator building
There are individual turbines for each of the reactor modules that are skid mounted and standard models currently available or several reactor modules can provide of steam one turbine-generator.

Electric power systems
Electric power system for pilot plant of 100 MW capacity is underway.

8. Design and Licensing Status

The Rosatom Scientific and Technical Council convened on 15 June 2006 approved the development of the technical design of experimental industrial power unit based on the SVBR-100. Siting licence is received to current time at Dimitrovgrad, in the region of Ulyanovsk. Key reactor and reactor core research and development works have begun. A complete reactor and power plant design is expected to be completed by 2020, along with a preliminary safety report. A construction licence is also expected to be obtained. The trial unit is expected to be commissioned in early 2020s.

9. Plant Economics

Russian Federation built seven Alfa-class submarines, each powered by a compact 155 MW(th) LBE cooled reactor, and 80 reactor-years operational experience was acquired with these. Overnight capital cost for pilot plant is previously estimated as $4000-4500/kW and generating costs 4-5 c/kWh on 90% load factor. Technical and economical assessments have shown that a number of SVBR-100 modules can replace the 2nd, 3rd, and 4th unit of the Novovoronezh NPP (NVNPP) at half the capital cost of replacing the same power capacity. The SVBR-100 structural design includes the requirements of versatility in terms of the fuel used to enable fast transition to MOX and later to nitride fuel as well, to serve the basis for fuel self-sufficiency in the closed nuclear fuel cycle.

G4M (Gen4 Energy Inc., United States of America)

1. Introduction

The G4M power module is a low capacity, portable and self-contained reactor designed to deliver a 70 MW(th) power with lead–bismuth eutectic as the primary coolant. The G4M does not need a human operator to generate a capacity of 25 MW(e). It has a fuel cycle of 10 years without on-site refuelling. The G4M conceptual design is being developed by the Gen4 Module (G4M) Inc. formerly Hyperion Power Generation Inc. formed in 2007. The design aims for achieving the operations of the standalone small power module simple enough that only a limited number of personnel are required to keep it operating safely and reliably. The reactor is intended to be sealed at factory, sited underground and eventually returned to the factory for fuel recycling and refuelling after a useful life of 10 to 15 years. The G4M has the following attributes:

- Single-unit, sealed construction and underground siting;
- Inherent simplicity and compactness of the design enable mass production and ease of shipping in off-the-shelf containers;
- Modest size and design simplicity greatly reduce the financial investment and enhance system reliability;
- Low power output and passive backup coolant systems combine to provide a high degree of safety.

Figure 1 shows the G4M reactor module based on 25 MW(e) power plant.

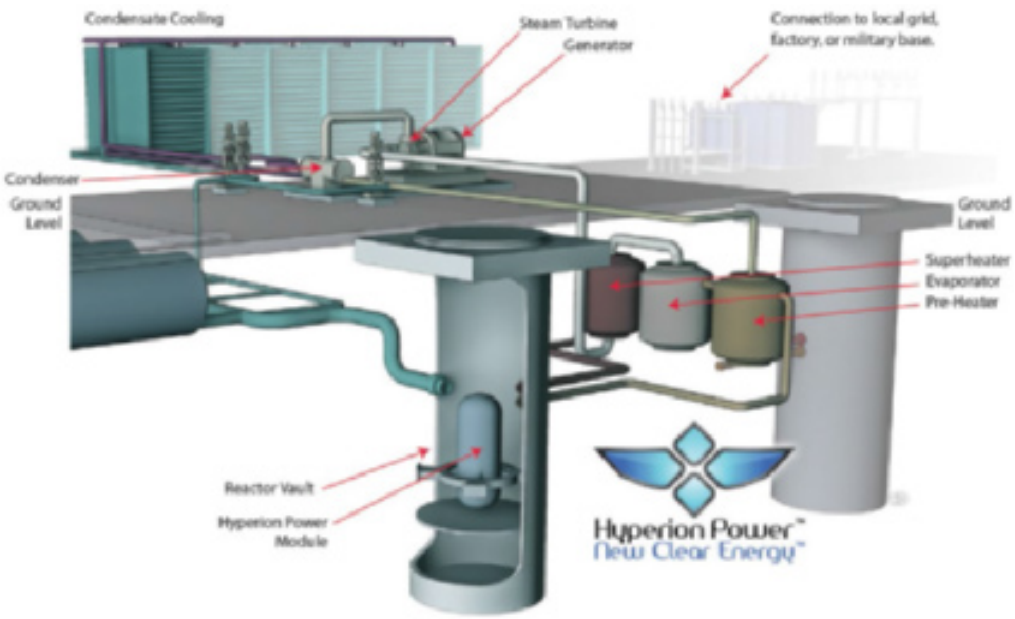


Figure 1: The G4M reactor module based on 25 MW(e) power plant (Reproduced courtesy of Gen4 Energy)

2. Target Application

The flexibility of G4M module to produce steam/electricity lends itself to applications ranging from supplying process heat for heavy industry to electricity for consumers electrical power applications. The design organization identified three main markets for which its

technology has a potential to be deployed: mining/oil and gas production, isolated and island communities and government facilities. Although G4M design can be located in any environment, the design is strategically best suited for remote locations or harsh climate environments.

3. Development Milestones

2007	Reactor design started
2008–2009	Engineering specifications
2010	Plans to build a G4M at Savannah river site in partnership with Hyperion Power Generation (HPG)
2012	Announced that they would not be pursuing the US Department of Energy's SMR licensing support programme
2015	Conceptual design completed

4. General Design Description

Design Philosophy

The G4M is designed to avoid the high construction costs and long construction times of the traditional large light water reactors and to provide power to remote areas. Another design objective of the G4M reactor design strategy is transportability and safety during that operation. The reactor concept is designed to eliminate or reduce the likelihood of accidents during transportation. This includes:

- Designed to eliminate the potential of criticality accidents;
- Elimination of any interaction of the reactor coolant with water (by choice of lead bismuth eutectic as a reactor coolant);
- Protection from fire and a highly reliable means of cooling on transport back to the factory.

The G4M power module cycle is shown in *Figure 2.*

Figure 2: The G4M power module cycle (Reproduced courtesy of Gen4 Energy)

Reactor Core

The reactor is approximately 1.5 m in diameter and 2.5 m in height, in which there are 24 subassemblies containing the fuel pins. The adopted fuel is uranium nitride (UN) pellets with 19.75% enrichment is contained in clad tubes made of HT-9. Quartz is used as a radial reflector. These high-temperature ceramic material pellets deter the ability to separate plutonium from spent fuel. The pin assembly is filled with liquid LBE to provide a high conductivity thermal bond between the fuel and cladding. The gap in the fuel pins is sized to preclude fuel clad mechanical interference throughout the core's lifetime. A plenum is located at one end, which serves both as a fission gas plenum and a repository for the LBE inside the pin as the fuel swells with burn-up. The fuel pellet stack is between two and three meters high, which includes neutron reflectors and the gas plenum. There are four types of sub-assembly configurations shown in *Figure 3*. The geometries of these sub-assemblies are dictated by the locations and sizes of the reserve shutdown cavity and control rods in the core array.

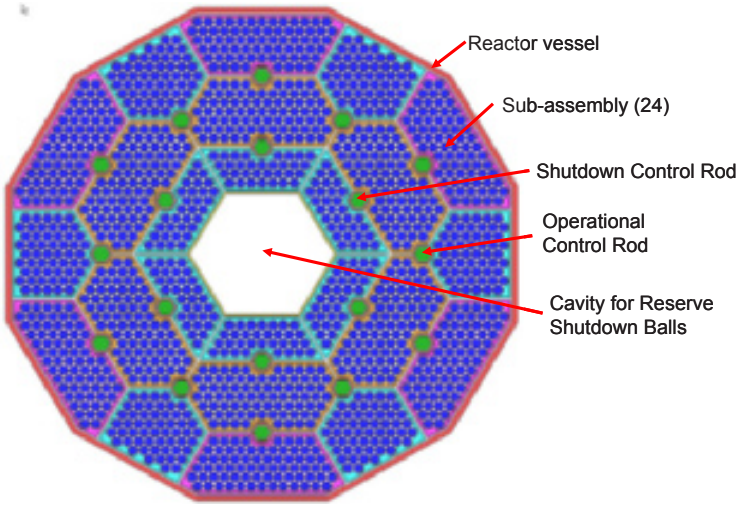


Figure 3: G4M core plan view (Reproduced courtesy of Gen4 Energy)

Reactivity Control

There are three (3) independent reactivity shut-down systems in the core: (1) a shutdown rod system composed of six (6) boron carbide (B_4C) rods; (2) a control rod system comprising 12 boron carbide (B_4C) rods; and (3) a reserve shutdown system consisting of a central cavity into which B_4C balls may be inserted. Each of the three systems can independently take the core to long-term cold shutdown. The rod shutdown and the ball shutdown systems perform this safety function automatically and instantaneously when triggered. Both the control rods and the spheres are inserted into dry wells in the core, which are hexagonally shaped thimbles. These thimbles penetrate the reactor vessel and are sealed from the primary coolant. Both systems can independently take the core to long term cold shutdown.

Reactor Pressure Vessel and Internals

The outer diameter of the reactor system, including the outer reflector and coolant down-comer, is limited to 1.6 m to be able to seal the reactor vessel system at the fabrication facility and transport it to the site in a conventional shipping cask. The total mass of the reactor vessel with fuel and coolant is 18 metric tonnes.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	Gen4 Energy Inc.
Country of origin	United States of America
Reactor type	Liquid metal cooled reactor
Electrical capacity (MW(e))	25
Thermal capacity (MW(th))	70
Expected capacity factor (%)	> 90
Design life (years)	5–15 (nominal 10) years
Plant footprint (m^2)	–
Coolant/moderator	Lead-bismuth
Primary circulation	Forced circulation
System pressure (MPa)	Low pressure operation
Core inlet/exit temperatures (^oC)	– / 500
Main reactivity control mechanism	Rod insertion and B_4C ball insertion
Power module height (m)	2.5
Power module wide (m)	1.5
RPV or module weight (metric ton)	18
Configuration of reactor coolant system	Integral pool type
Power conversion process	Indirect Rankine cycle
Fuel type/assembly array	Uranium nitride
Fuel assembly active length (m)	-
Number of fuel assemblies	24
Fuel enrichment (%)	19.75
Fuel burnup (GWd/ton)	–
Fuel cycle (years)	10
Cogeneration capability	Yes
Approach to engineered safety systems	Hybrid (Active + Passive)
Number of safety trains	2
Refuelling outage (days)	–
Distinguishing features	Transportable factory fuelled design, a fuel cycle of 10 years, under grade containment system.
Modules per plant	1
Target construction duration (months)	–
Seismic design	–
Core damage frequency (per reactor-year)	–
Design Status	Conceptual design

Reactor Coolant System

The core coolant is LBE, with a mixed mean exit temperature of 500°C. This temperature limits the cladding temperature, so that maximum cladding creep over the 10 year lifetime of the reactor is less than 1%. A solid phase oxygen control system used to control the oxygen level in the coolant to maintain a protective coating on the clad surfaces, limiting corrosion in the clad to acceptable levels. The pitch-to-diameter ratio of the fuel pins in the core was set to limit the coolant velocity to 1.9 m/s, to prevent erosion of the protective coating on the fuel

pin clad surfaces. The use of LBE liquid metal coolant enhances the safety and radically reduces the potential energy available for dispersion of radioactivity in a worst-case accident.

- Potential radioactivity releases from vaporization of the coolant are eliminated because of the high boiling temperature of LBE;
- Potential radioactivity releases due to pressurized release of the coolant are eliminated as LBE is at near-atmospheric pressure;
- LBE is non-reactive with air, water, fuel and metal, thus rapid exothermic reactions with associated radioactivity releases are eliminated. Additionally, reactions that produce flammable hydrogen are eliminated.

The heat exchanger placed on the top of the sealed vessel transfers the thermal power generated in the reactor core to the secondary heat pipes that are larger and longer so they can reach grade level and subsequently transfer the thermal power to a working fluid used as a hear supply or used to drive rotating machinery to produce electricity.

5. Safety Features

The several safety features of the G4M module result in a safe, reliable and proliferation-resistant method of generating power, that is analysed thoroughly for all radioactive related releases, natural threats such as seismic, tornado/missile, etc. Among the safety principles that G4M power module has adhered to include:

- Natural circulation of the primary coolant to remove the heat from the core in the event of a loss-of-flow incident;
- Using highly reliable passive safety features to decrease the probability of failure;
- Using the principle of defense-in-depth to further decrease the frequency of accidents that could impact public safety, including using multiple containment to prevent any release of radioactive materials. This principle also applies to physical security as an objective to prevent malevolent intrusion from any outside threat.

Decay heat removal system

During operational shutdown, decay heat is removed from the G4M by two methods. During operational shutdowns, decay heat is removed from the core through the normal coolant pathway by dumping steam to the condenser. The second backup decay heat removal system provides natural circulation of LBE through a fixed bypass path in the core. Water from an emergency cooling tank is gravity-sprayed onto the exterior surface of the G4M module, and heat is removed by passive vaporization of water. The backup decay heat removal system can provide adequate cooling for up to 14 days without any power or operator action.

Containment system

The G4M module is sited in an underground containment vault to provide isolation from the environment, prevent intrusion or tampering and avoid accidents from extreme external events. The inner vessel confines the hydrogen gas and provides containment for all radioactive materials. The containment vessel is designed for double confinement. The hardened containment vault is designed to protect against extreme external events and man-made events. Any radioactivity release from the G4M module, however unlikely, is contained in the vault. The underground vault eliminates or minimizes the potential impact of natural disasters such as floods, hurricanes and tornados by preventing any contact of wind, water or debris with the reactor. The vault prevents the possibility of intrusion or tampering, as well as minimizing the impact of security threat scenarios such as an airplane impact or an explosive device. Because the G4M module is factory sealed, never opened on site, and located in an

underground vault, the potential for any radioactive contamination on the site is virtually eliminated.

6. Plant safety and Operational Performances

The reactor operates at an optimum temperature of 550°C, picked as the goal for the Generation IV reactors. At 550°C, the dissociation pressure for the hydrogen above the hydride is approximately eight atmospheres, which permits easy transportation of the gas without presenting significant high-pressure risk. The temperature-driven mobility of the hydrogen contained in the hydride can change the moderation, and therefore the reactor criticality, making the reactor self-regulating. The hydrogen forced out of the core during any over-temperature excursion reduces the neutron energy moderation necessary for nuclear criticality. The G4M power module is inherently fail-safe, since any temperature increase from excess activity immediately reduces the criticality parameters and thus the power production. The consequent power reduction causes the temperature to decrease and that temperature decrease eventually reverses the process, resulting in relaxation oscillations that quickly damp out to steady-state operation.

7. Instrumentation and Control systems

The complex arrays of detectors, analysers and control systems responsible for the safety and stability of conventional nuclear power reactors have been superseded by the fundamental science and properties of the active materials.

8. Plant Arrangement

The G4M plant layout diagram is shown in *Figure 4*, below.

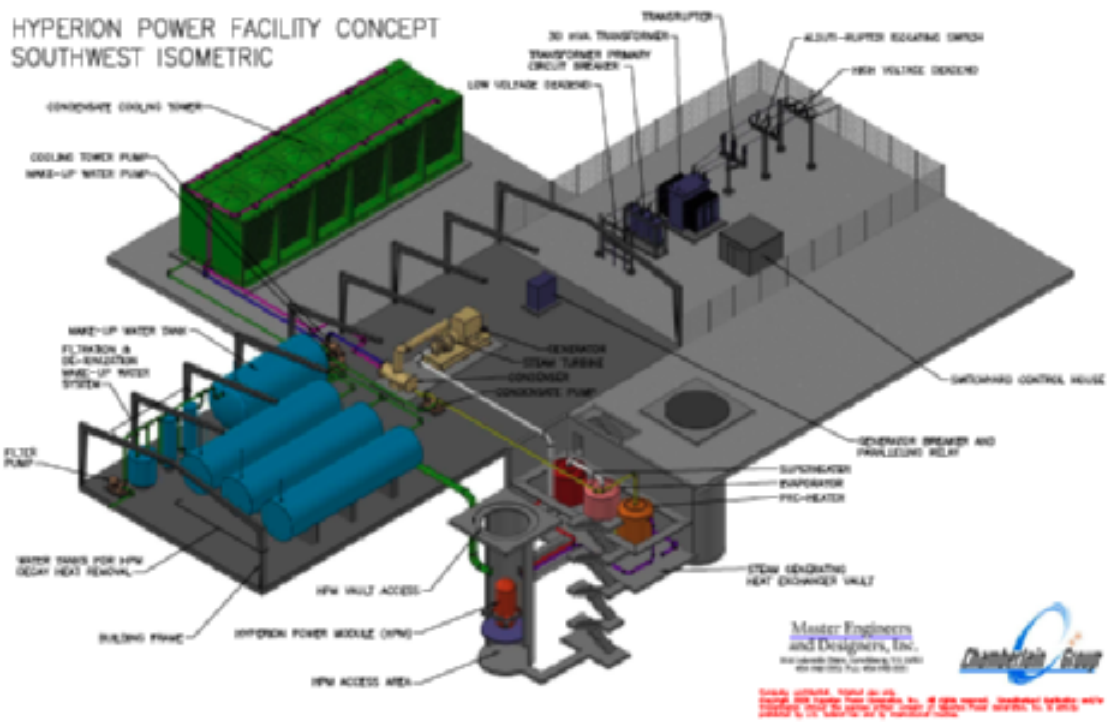


Figure 4: The G4M plant layout diagram (Reproduced courtesy of Gen4 Energy)

9. Design and Licensing Status

Gen4 Energy presented its conceptual design to the U.S. NRC, and according to the company, to appropriate licensing authorities in other countries. The company believes that its initial clients will be best served by licensing the design either in the U.S., Canada or the U.K. Although the first unit is likely to be deployed at the Savana River Site, Gen4 Energy is also in talk for potential deployment of the G4M in Canada. Gen4 Energy announced in April 2012 that they would not be pursuing the US Department of Energy's small modular reactor licensing support programme because they concluded that, use of well-known light water reactor technology of 45 to 300 MW(e) intended for deployment in the United States had a much higher probability of success given the (funding opportunity announcement's) stated maximum of two awards.

10. Plant Economics

Power generation is expected to be economically better than the large-scale nuclear power plants and cheaper, from a broad point of view, than coal, gas, fuel oil, etc.

EM^2 (General Atomics, United States of America)

1. Introduction

Energy Multiplier Module (EM^2) is a helium-cooled fast reactor with a core outlet temperature of 850°C. It is designed as a modular, grid-capable power source with a net unit output of 265 MW(e). An elevation view of an EM^2 modular building element is shown in *Figure 1*. The reactor employs a "convert and burn" core design which converts fertile isotopes to fissile and burns them in situ over a 30-year core life. EM^2 employs a direct closed-cycle gas turbine power conversion unit (PCU) with an R-245fa Rankine bottoming cycle for 53% net power conversion efficiency assuming evaporative cooling. EM^2 is multi-fuel capable, but the presented case assumes low-enriched uranium (LEU) with depleted uranium (DU) fertile fuel.

Figure 1: Elevation view of EM^2 modular building element employs two modules on a single seismically isolated platform (Reproduced courtesy of General Atomics)

2. Target Applications

The EM^2 is being developed for the electricity generation and high temperature use.

3. Development Milestones

Phase I	High risk R&D
	Conceptual design of reactor core and power plant
	Fuel design and demonstration
	High temperature material development
Phase II	Demonstration/prototype plant
	Qualification of fuel
	Qualification of plant operation
Phase III	First-of-a-kind plant
	Demonstration of commercial operation

4. General Design Description

Design Philosophy

EM^2 design initiative is to develop a new nuclear plant to address the following goals for enhancing the likelihood of commercial success:

- Economic parity with fossil fuel generation in the U.S.
- Improved siting flexibility via dry cooling and site accessibility
- Passive safety for sustained protection during long-term station blackout and other severe accidents
- Improved fuel resource utilization, reduced nuclear waste and high proliferation-resistance

Reactor Core

The core is supported by the support floor through the core barrel, which attaches to the vessel below the cross-duct. The upper carbon-composite (C-C) heat shield protects the top head elements from the hot helium. The vessel is internally insulated with silica/alumina fibrous insulation retained with a C-C cover plate. In order to achieve high fuel utilization, the EM^2 core utilizes the "convert and burn" concept, in which the core is divided into two core sections: fissile and fertile. The fissile section is the "critical" section at beginning of life (BOL). It contains ~14.5% LEU to sustain the chain reaction and provide excess neutrons to convert DU fertile to fissile material. The average enrichment of the total active core is 7.7%. The reflector consists of an inner section of zirconium-alloy blocks and an outer section of graphite blocks.

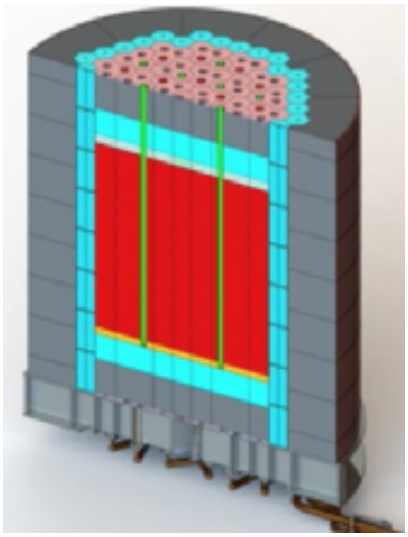


Figure 2: Reactor core arrangement (Reproduced courtesy of General Atomics)

The basic building block of the advanced EM^2 fuel system is the hex-assembly, of which there are 85 in the core. Eighty-one hex-assemblies are joined into 27 tri-bundles and 4 remain as individual hex assemblies. The tri-bundle is located between separate upper and lower reflector assemblies. It has a bottom alignment grid, an upper manifold assembly, and one intermediate spacer grids. The fuel for EM^2 is contained in cylindrical fuel rods arranged in a triangular pitch. Due to the high operating temperatures and long fuel cycle, all tri-bundle structural components and cladding are made of SiC-SiC.

Fuel Characteristics

Uranium carbide (UC) is used to meet the high uranium loading requirement; has a very high thermal conductivity; is compatible with the SiC-SiC cladding; and has a suitably high melting point. Each annular fuel pellet is a sintered "sphere-pac" with a specified interstitial and internal distributed porosity to allow for faster migration of volatile fission products. SiC-SiC is especially attractive due to its stability under long term irradiation as demonstrated in a multi-year irradiation campaign. Both the fuel and cladding materials meet design criteria temperature limits for both normal operations and accident conditions.

MAJOR TECHNICAL PARAMETERS:	
Parameter	**Value**
Technology Developer	General Atomics
Country of Origin	USA
Reactor Type	Modular High Temperature Gas-cooled Fast Reactor
Electrical Capacity (MW(e))	265
Thermal Capacity (MW(th))	500
Expected Capacity Factor (%)	90
Design Life (years)	60
Plant Footprint (m^2)	~90000 (4 modules)
Coolant/Moderator	Helium / None
Primary Circulation	Forced circulation
System Pressure (MPa)	13.3 (peak)
Main Reactivity Control Mechanism	Control rod drive mechanism
RPV Height (m)	12.5
RPV Diameter (m)	4.6
Coolant Temperature, Core Outlet (°C)	850
Coolant Temperature, Core Inlet (°C)	550
Integral Design	No
Power Conversion Process	Direct Brayton top cycle and Rankine bottom cycle
High-Temp Process Heat	No
Low-Temp Process Heat	No
Cogeneration Capability	Yes
Design Configured for Process Heat Applications	No
Passive Safety Features	Direct reactor auxiliary cooling system (DRACS)
Active Safety Features	Control rods and shutdown rods
Fuel Type/Assembly Array	UC pellet/ hexagon
Fuel Enrichment (%)	~14.5
Fuel Burnup (GWd/ton)	~130
Fuel Cycle (months)	~360
Number of Safety Trains	N/A
Emergency Safety Systems	Control rods and shutdown rods
Residual Heat Removal Systems	Passive
Refuelling Outage (days)	14
Distinguishing Features	Silicon carbide composite cladding and fission gas vent system
Modules per Plant	4
Estimated Construction Schedule (months)	42
Seismic design (g)	N/A
Predicted large release frequency	N/A
Design Status	Conceptual design

Fuel Handling
The core is accessed by a refuelling machine from the maintenance hall floor. An articulated arm extends through the containment and reactor vessel penetration to select and withdraw a tri-bundle assembly and load it into a sealed, air-cooled storage container. The container is moved to the end of the maintenance hall where it is lowered into the fuel storage facility. This facility has the capacity for 60 years of operation. The spent fuel is cooled within the sealed containers by passive natural convection of air. No water or active cooling is required.

Reactivity Control
Reactivity control is provided by the 18 control rods and 12 shutdown rods. Both control rod system and shutdown rod system each have sufficient negative reactivity to render the core cold subcritical. The control and shutdown drives are located at the top of the reactor vessel. The control rod drives utilize a ball-screw drive while the shutdown rods use linear motors.

Reactor Pressure Vessel and Internals
The reactor pressure vessel (RPV) is constructed from welded ring-forgings or rolled plate. The bottom head contains three large penetrations for the two cross-vessels and a flanged hatch near the bottom. The top head has the penetration for refuelling access and control elements. The RPV has no external insulation, but is internally insulated. This insulation maintains the vessel well below 371°C during normal operation and design basis accidents, which allows the use of SA-533 grade B material.

Power Conversion Unit
The power conversion is based on a combined cycle with a direct helium Brayton cycle and a Rankine bottoming cycle. The helium Brayton cycle is located in the PCU vessel, while Rankine cycle is located outside the reactor building. The Brayton cycle incorporates the turbo-compressor (T/C) and generator which are mounted on an in-line vertical shaft suspended by active magnetic bearings as shown in *Figure 3*. The cycle also incorporates two heat exchangers (HX), a recuperator and a precooler. The generator uses a permanent magnet (PM) rotor to eliminate I^2R losses associated with a wound rotor and exciter.

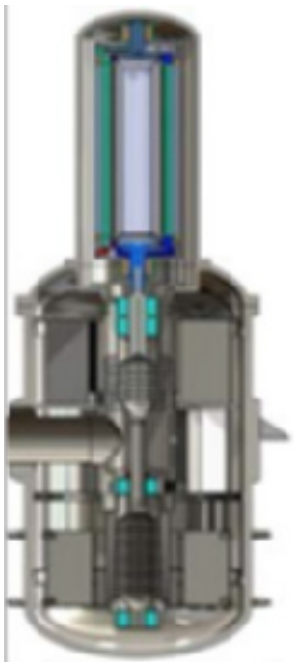


Figure 3: EM ^{2} PCU enclosed Brayton top cycle (Reproduced courtesy of General Atomics)

5. Plant Safety Features

Description of Safety Concept
The EM^2 safety design uses a defence-in-depth approach which employs three successive, encompassing barriers against the release of radionuclides. Each barrier relies ultimately on passive means for protection of its integrity for normal and abnormal operation. The first barrier is the SiC-SiC fuel cladding. The second barrier is the primary vessel system, which encompasses the reactor, PCU, and DRACS. The third barrier is the free-standing, below-grade containment. Because the fuel is vented, the fission product vent system (FPVS) is an extension of the first barrier.

Fuel Cladding
The SiC-based composite cladding is a superior material being developed for accident tolerant fuel (ATF). SiC-SiC maintains its strength up to 1700ºC and retains greater than 75% of its strength at 2000 ºC.

Fission Product Vent System
The FPVS protects fuel cladding by venting gaseous fission products to high temperature adsorber in the containment. The venting effectively reduces the volatile fission product inventory in the fuel rods and reduces the source term for accidents.

Direct Reactor Auxiliary Cooling System
The DARCS safely removes core afterheat during normal shutdown and accident conditions when the PCU is not available. It provides controlled core heat removal during an anticipated transient without scram (ATWS) and active core heat removal during special maintenance conditions characterized by need for low temperatures and/or low helium pressures.

Containment System
The primary heat transport system (PHTS) is enclosed by a sealed, below-grade containment, which is divided into three connected chambers with structural ligaments around the reactor chamber that also serve as shielding to all access to the two side chambers. The containment is hermetically sealed with an inert (argon) atmosphere at ~20 psig. The peak pressure rating is 90 psig. The design leakage rate is less than 0.2% per day.

Protection from External Events
Protection from external events is provided by locating the reactor containments and the spent fuel storage facility below grade as shown in *Figure 4*. The roof of the maintenance hall above the reactors and spent fuel storage is an aircraft crash shield per NRC regulations. The containment and reactor auxiliary building are mounted on a common seismic isolation platform similar to that used for large building in seismic areas.

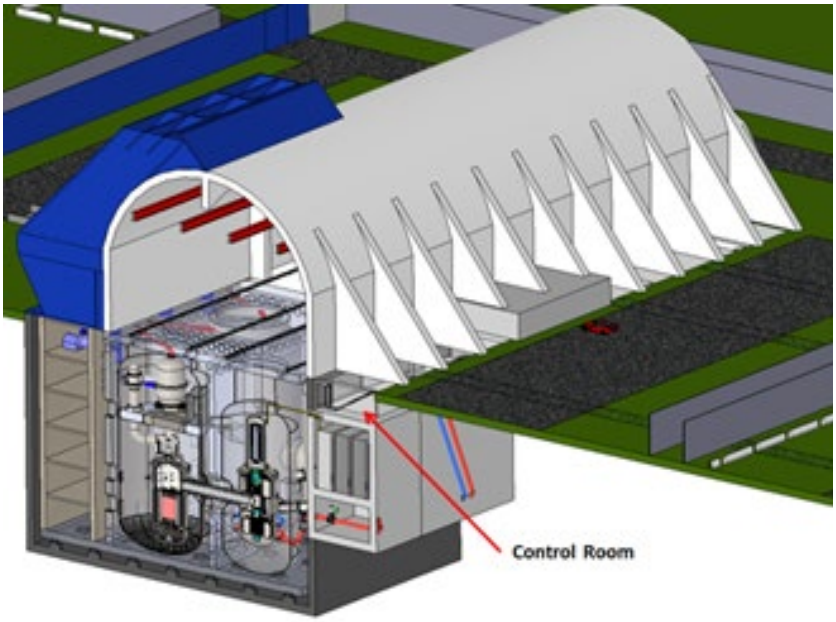


Figure 4: Vertical cross-section of EM^2 plant showing below-grade containment and crash shield. (Reproduced courtesy of General Atomics)

6. Plant Operational Performances

Unlike large power reactors, each EM^2 module utilizes a unique, non-synchronous, variable speed generator with a frequency inverter and generator load commutator to follow load demand. In the automatic load following mode, the generator speed is set by modulating the generator speed. A field-oriented control algorithm in the frequency converter controls the generator torque that decreases or increases the generator speed, which in turn determined helium flow. The ability to control the turbo-generator speed through field control replaces traditional mechanical control elements with digital electronic control. An advantage of the variable speed control is that it maintains primary system structures at near constant temperature with load so that rapid load following should be possible.

7. Instrumentation and Control Systems

EM^2 deploys advanced sensors for high operating temperature condition such as solid-state and silicon carbide (SiC) neutron flux monitors. The plant control including startup, operation and shutdown is conducted through integrated control system actions, which regulate reactor power and turbo-machinery to respond to the plant transients. Plant control functions are performed by the power control and the process component control systems. The power control system includes control rod drive mechanism (CRDM) and reactor coolant system (RCS). The process component control system includes a non-synchronous, variable speed generator and a frequency inverter.

8. Plant Arrangement

The baseline EM^2 plant is composed of four 265 MW(e) modules for a combine net power of 1,060 MW(e) to a utility grid for evaporative cooling and 960 MW(e) net for dry-cooling. Each module consists of a complete powertrain from reactor to heat rejection such that the modules can be built sequentially and operated independently. *Figure 5* shows the plant layout, which covers 9.3 hectares (23 acres) not including the switchyard. The maintenance hall floor is at grade level, and the roof serves as a protective shield structure. The maintenance hall serves all four reactors.

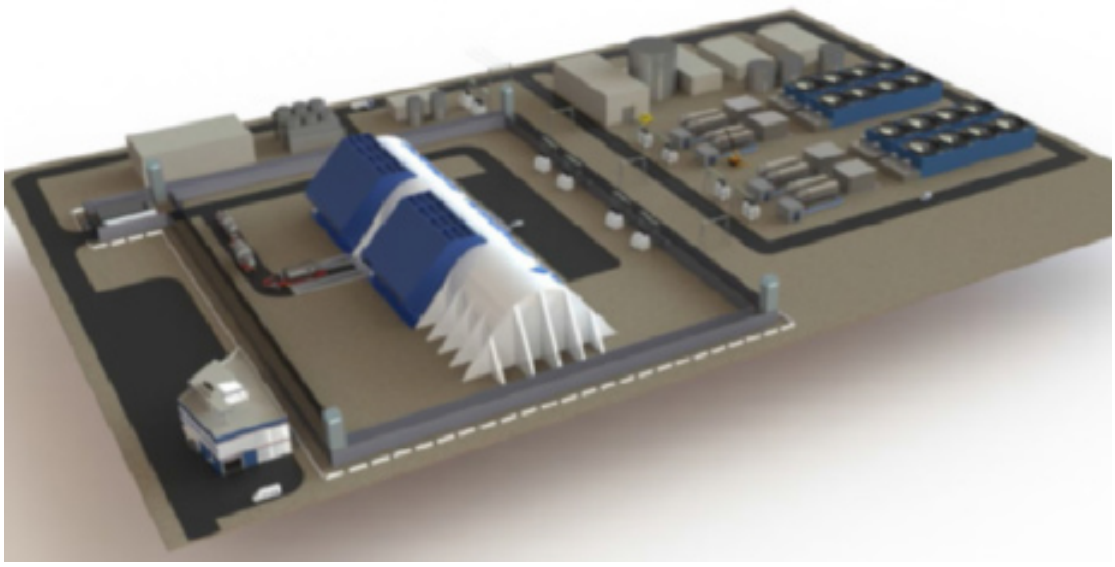


Figure 5: One gigawatt EM^2 plant layout on 9 hectares of land (Reproduced courtesy of General Atomics)

Reactor Building

The reactor building as shown in *Figure 1* is divided into two sets of two module separated by the electrical distribution building and access entry. Two reactor modules with individual containment assemblies are mounted on a seismic isolation platform. The reactor auxiliary building is also mounted on the platform. The containment structure is suspended from an approximate mid-plane support frame that also supports the primary system. Access to the reactor, PCU and DRACS units is from the maintenance floor at grade level.

Balance of Plant

The primary system is enclosed within a sealed, containment as shown in *Figure 6* with two chambers connected by a duct. The primary system includes the reactor and PCU whose respective vessels are connected by a concentric cross-duct. The reactor vessel is also connected to two DRACS systems (only one shown in *Figure 6)*. Natural circulation paths are provided by the vertical concentric cross-ducts to helium-to-water heat exchangers. The maintenance circulators, which are normally valve-off, are used for low pressure maintenance conditions.

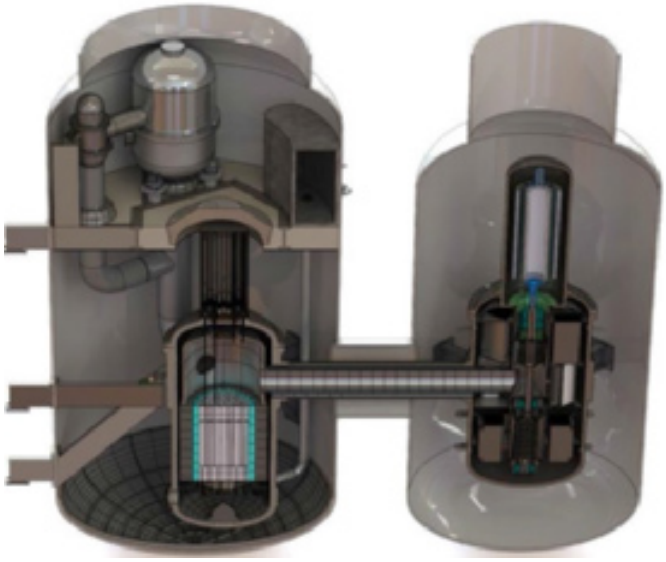


Figure 6: EM^2 primary system enclosed in two-chamber sealed containment (Reproduced courtesy of General Atomics)

9. Design and Licensing Status

The EM^2 prototype plant will be licensed using the two-step 10 CFR Part 50 and follow-on commercial plants will be licensed under the one-step 10 CFR Part 52 process.

10. Plant Economics

The target equilibrium plant overnight capital unit cost of a four module installation is about $4,300/kWe (2014$) plus and additional first core cost of $240/kWe. The equilibrium levelized cost of electricity is about $67/MWh assuming a weighted average cost (WAC) of capital of 8.4%.

For further information: http://www.ga.com

MOLTEN SALT
SMALL MODULAR REACTORS

Integral Molten Salt Reactor (Terrestrial Energy, Canada)

1. Introduction

The Integral Molten Salt Reactor – 400 Megawatts-thermal (IMSR400) is a small modular molten salt fuelled reactor [1]. It features a completely sealed reactor vessel with integrated pumps, heat exchangers and shutdown rods. This vessel, called the IMSR core-unit, is replaced completely as a single unit at the end of its useful service life (nominally 7 years). This allows factory production levels of quality control and economy, while avoiding any need to open and service the reactor vessel at the power plant site.

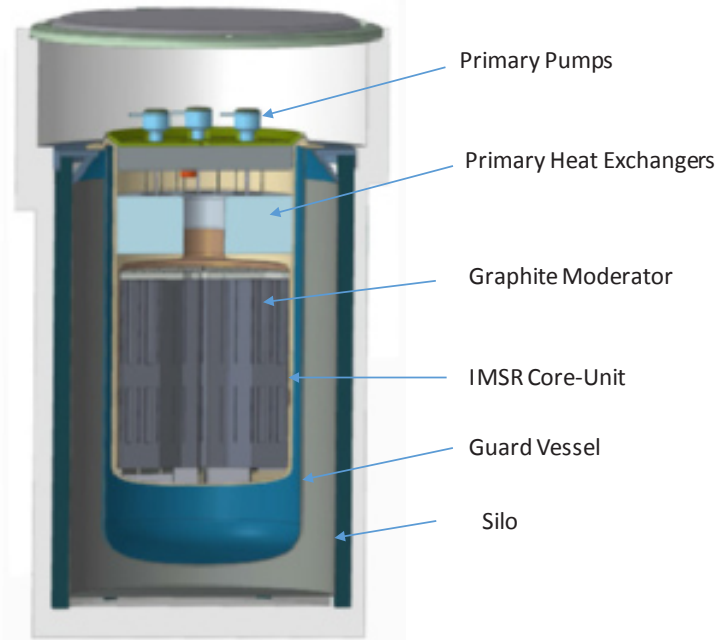


Figure 1: Core-unit and guard vessel in one of the two reactor silos (Reproduced courtesy of Terrestrial Energy)

For ultimate safety, there is no dependence on operator intervention, powered mechanical components, coolant injection or their support systems such as electricity supply or instrument air in dealing with upset conditions. This is achieved through a combination of design features:

- The inert, stable properties of the salt;
- An inherently stable nuclear core;
- Fully passive backup core and containment cooling systems;
- An integral reactor architecture.

2. Target Application

The IMSR plant is designed to accommodate various load users, from baseload to load-following. Featuring a simple modular and replaceable core-unit, very high reliability ratings are targeted. The IMSR has been specifically designed for factory fabrication. Nuclear components are small and road-transportable. The IMSR core-unit is designed for a short service lifetime, which allows dedicated factory lines to produce the units semi-automatically, similar to aircraft jet engine production lines for example.

3. Development Milestones

2015	Concept design completed
2016	Start of pre-licencing vendor design review (by Canadian nuclear regulator)
2016	Start of basic engineering phase
Early 2020's	Secure necessary licenses
Early 2020's	Plan construction of a first full-scale NPP in Canada

4. General Design Description

Design Philosophy

The design is based heavily on the intensive molten salt reactor project of Oak Ridge National Laboratory (ORNL) in the 1950's - 1970's. In this project, extensive R&D was done and molten salt reactor materials and equipment were developed. This included such items as suitable graphite to act as neutron moderator, drain tanks for the salts, hot cells for housing the equipment, pumps and heat exchangers. The ORNL program culminated in the construction and successful operation of the Molten Salt Reactor Experiment (MSRE). The MSRE employed a molten fluoride fuel salt, which was also the primary coolant. This fuel-coolant mixture was circulated between a critical graphite moderated core and external heat exchangers. The IMSR is essentially a re-configured, scaled-up version of the MSRE. This minimizes required research and development (R&D). The IMSR has also been influenced by a more recent ORNL design, the small modular advanced high temperature reactor (SmAHTR) which was to use solid fuel with fluoride salt coolant, but had an "integral" architecture where all primary pumps, heat exchangers and control rods are integrated inside the sealed reactor vessel.

Power Conversion System

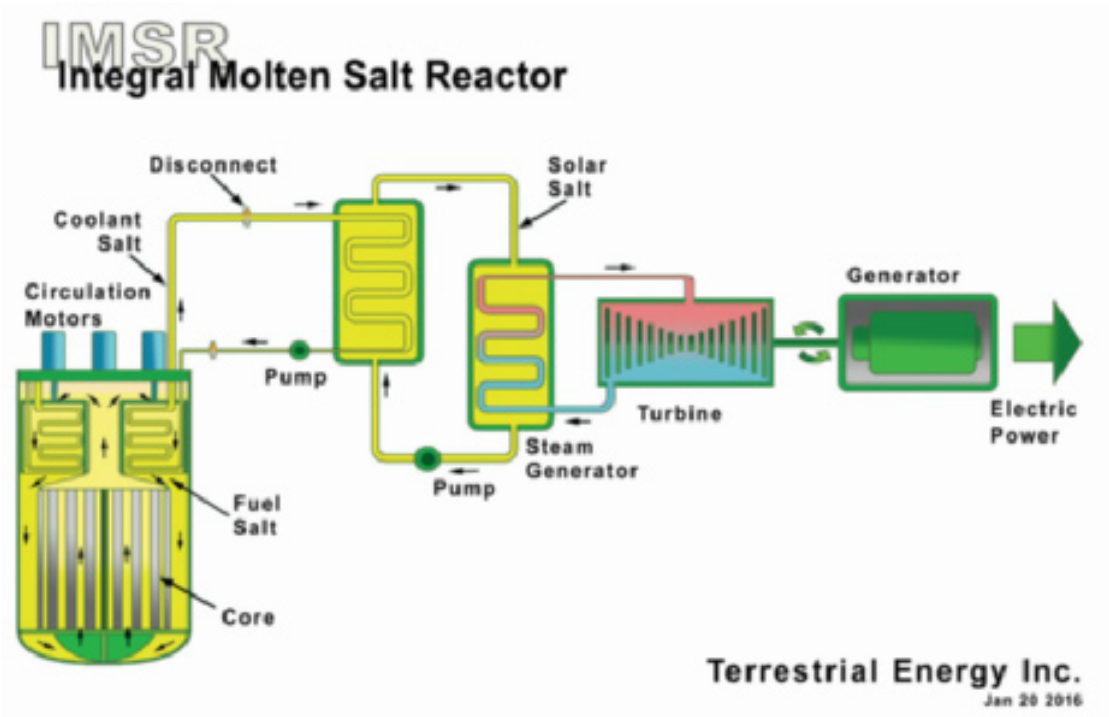


Figure 2: IMSR400 main heat transport path for power generation (Reproduced courtesy of Terrestrial Energy)

The IMSR uses molten fluoride salt, a highly stable, inert liquid with robust coolant properties and high intrinsic radionuclide retention properties, for its primary fuel salt. A secondary, coolant salt loop, also using a fluoride salt (but without fuel), transfers heat away from the primary heat exchangers integrated inside the core-unit. As shown in the *Figure 2*, the coolant salt loop, in turn, transfers its heat load to a solar salt loop, which is pumped out of the nuclear island to a separate building where it either heats steam generators that generate superheated steam for power generation or is used for process heat applications.

Reactor and Core-unit

The IMSR is an integral nuclear reactor design. This means that the reactor core, pumps, heat exchangers and control rods are all mounted inside a single vessel, which is referred to as the IMSR core-unit. This core-unit is manufactured in a controlled factory environment and then brought to the reactor power plant site where, following final assembly, it is lowered into a surrounding guard (containment) vessel which itself sits in a below grade reactor silo. There, the core-unit is connected to secondary piping, which contains a non-radioactive coolant salt. For optimal safety and containment, this coolant salt is used as an inert, low pressure intermediate loop between the critical reactor circuit and the steam power generation circuit. The steam circuit powers a conventional, off-the-shelf industrial steam turbine for power generation and/or industrial steam production, depending on the required application. Alternately, some or all of the hot molten salt may be sent directly to a process heat application.

Fuel Characteristics

Being a liquid fuel reactor, there are no fuel elements in the reactor core. The fuel may consist of low enriched uranium fluoride, plutonium fluoride, thorium fluoride, or any mixture of these. The first of a kind IMSR400 however, will utilize a once-through, low enriched uranium fuel cycle as this is the simplest option. This fuel salt is diluted with coolant salt, consisting of fluorides such as sodium fluoride, beryllium fluoride and/or lithium fluoride. Together this mixture constitutes both fuel and primary coolant. The fuel-coolant mixture is pumped between a critical, graphite moderated (thermal spectrum) core, and then through the integral heat exchangers to transfer its heat to the secondary, coolant salt loop. The latter loop consists of bare diluent salts (without fuel salt added), that in turn transfers its heat to another intermediate salt loop. This second intermediate salt loop or “third loop” improves safety by adding another barrier between the radioactive primary inventory and the steam turbine. This third loop uses a nitrate salt, for its lower melting point (to avoid steam generator freezing) and compatibility with steam (in the event of a leak). Finally, the nitrate salt heated steam generator produces steam for process heat or to power a steam turbine-generator set.

Fuel Handling and Core-unit Replacement Approach

The new or “fresh” fuel is separately brought to the power plant site as a solid, where it is melted and added to the IMSR core-unit. This allows the IMSR to operate with online fuelling. Additionally, and unlike solid-fuel reactors, there is no need to remove a proportion of old fuel during makeup fuelling. All of the fuel stays inside the closed IMSR core-unit during the entire power operations period of the core-unit. The volume of small amounts of additional “makeup” fuel salt is simply accommodated in the upper gas plenum.

Unlike other power reactor systems, the IMSR core-unit needs never be opened at the power plant site, either during start-up fuelling or during refuelling. After ~7 years of power operation, the core-unit is shut down and after a cool-down period, the used fuel charge is pumped out to robust holding tanks located inside the containment.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	Terrestrial Energy
Country of origin	Canada
Reactor type	Molten Salt Reactor
Electrical capacity (MW(e))	185 – 192
Thermal capacity (MW(th))	400
Expected capacity factor (%)	90
Design life (years)	60
Coolant/moderator	Fluoride fuel salt and graphite moderator
Primary circulation	Forced circulation: 5400 kg/s
System pressure (MPa(a))	< 0.4 (hydrostatic)
Core inlet temperatures (^{o}C)	625 – 660
Core exit temperatures (^{o}C)	670 – 700
Mean temperature rise across core (^{o}C)	75
Main reactivity control mechanism	Short term: Negative temperature coefficient; inherent Long term: Online liquid fuel additions
RPV inner diameter (m)	3.5
RPV height (m)	7.0
RPV mass (t)	170 (Transport weight)
Power conversion process	Supercritical steam turbine; two intermediate salt loops.
Cogeneration / Process Heat Capabilities	District heating, hydrogen production, liquid fuel production, Industrial co-gen, ammonia production, mineral resource extraction, petrochemical refining
Passive Safety Features:	negative temperature coefficients, large heat capacity, heat removal by natural means; flow driven control rods
Moderator type	Graphite
Fuel enrichment (%)	Depends on type of cycle, first of a kind to be low enriched uranium
Fuel cycle (months)	84 months before core-unit replacement
Residual Heat Removal System	based on heat large capacity of salt and graphite, heat loss through the core-unit; guard vessel and surrounding air
Refuelling outage (days)	Not applicable; second unit can be operated
Distinguishing features	completely sealed reactor vessel with integrated pumps, heat exchangers and shutdown rods; core-unit is replaced completely as a single unit at the end of its useful service
Modules per plant	Dependent on market and application
Target construction duration (months)	Factory manufactured units will reduce on-site construction
Seismic design	~0.3g (to be determined in detailed design phase)
Core damage frequency (per reactor-year)	Not applicable
Design Status	Conceptual design compete – Basic engineering in progress

The spent, empty core-unit is then allowed to cool down for several more years, at which point its radioactivity has diminished sufficiently to permit its safe removal from its containment silo. The IMSR400 is a single unit plant, however two core-unit silos are used, to allow switch-loading of the silos: this allows a long period of cool-down for the spent silo, while another is connected by switching over the secondary, coolant salt lines to the new unit. After this long cool-down period, the spent, empty core-unit is then moved into a long-term storage silo where it can stay for an extended period. Beyond this period the core-unit can be shipped to a central facility, to be recycled or prepared for geological sequestration. Similarly the separately stored spent fuel charge can also be shipped to a central facility to recover the fuel or for conversion to a form for sequestration. This sealed, integral reactor architecture and once-through, "burner" fuel cycle mode allows the actual power plant site to always operate in a clean, simple environment without risk of facility contamination.

The IMSR has intrinsic, design, and procedural features that resist proliferation of fissile materials for weapons production

Cooling System

The unique cooling system is based on heat capacity and heat loss – which are immutable. Heat capacity is due to the thermal mass of the fuel salt, vessel metal, and graphite. Heat loss occurs as the reactor vessel is not insulated. Short term cooling is provided by the low power density core-unit, and the internal natural circulation capability of the fluoride fuel salt, resulting in a large capacity to absorb transient and decay heat generation.

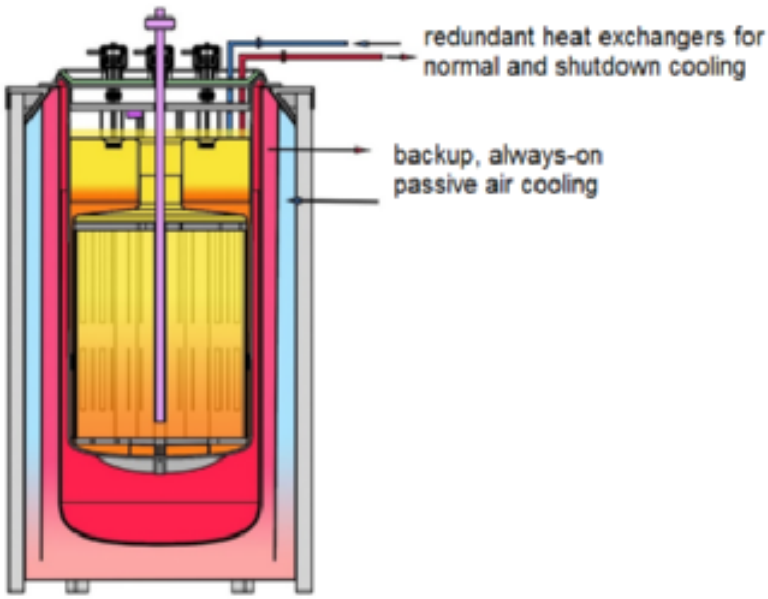


Figure 3: The heat loss approach for the IMSR core-unit
(Reproduced courtesy of Terrestrial Energy)

Longer term cooling is provided by heat loss from the uninsulated reactor vessel which itself is enveloped by a guard vessel. This guard vessel is a closed vessel that envelops the core-unit, providing containment and cooling through its vessel wall. Overheating of the core-unit will cause the core-unit to heat up and increase heat loss from the core-unit, in turn increasing heat transfer, via thermal radiation, to the guard vessel. The guard vessel in turn is surrounded by a robust air cooling jacket. This cooling jacket will provide long term cooling. The cooling jacket operates at atmospheric pressure so will continue to cool in the event of a leak or damage to the jacket. This is illustrated in *Figure 3*.

Reactivity Control
Reactor criticality control is assured through a negative temperature feedback made possible by molten salt fuel. This negative temperature feedback assures reactor safety on overheating, even with loss of all control systems. Molten salt fuel does not degrade by heat or radiation, so there are effectively no reactor power limits to the fuel.

For the control function, redundant, shutdown rods are also integrated into the IMSR core-unit. These shutdown rods will shut down the reactor upon loss of forced circulation and will also insert upon loss of power. Another backup is provided in the form of meltable cans, filled with a liquid neutron absorbing material that will shut down the reactor on overheating.

5. Safety Features

The basic design approach to safety in the IMSR is to achieve an inherent, walk-away safe nuclear plant. No operator action, electricity, or externally-powered mechanical components are needed to assure the most basic safety functions:

- Control: failure of control systems or reactivity insertion events only leads to reactor stabilization at a slightly higher temperature.
- Cool: inherent heat sinks are available initially to absorb transient and decay heat, with heat losses providing long term cooling for both core and containment.
- Contain: the fluoride salts are chemically stable, bind radioactive fission products to the salt, and have extremely high boiling points. Multiple engineered barriers are provided as backup to this inherent containment. Partial entrenchment of the reactor, combined with thick concrete and steel shielding, provides resistance to external events such as earthquakes, explosions, and aircraft crash.

An important part of the IMSR safety philosophy is to start by removing drivers that push radioactive material into the environment. Specifically the reactor always operates at low pressure due to the inert, low-volatile fuel-coolant mixture and the absence of water or steam in the reactor. This approach completely eliminates stored energy, both physical and chemical, from the reactor system. The IMSR further augments this high level of inherent, physics-based safety with its integrated, pipe-less, fail-safe systems architecture.

The IMSR does not depend on depressurizing the reactor or bringing coolant to the reactor. All required control and heat sink functions are already present where they are needed – in and directly around the IMSR core-unit. As such the IMSR completely eliminates any dependence on support systems, valves, pumps, and operator actions. This is the case in both the short term and the long term. To make this possible, the IMSR designers have combined molten salt reactor technology with integral reactor design and a unique cooling system.

Fission Product Retention
The main barrier for fission product release is the fluoride salt that is chemically stable and that bind radioactive fission products to the salt. The core-unit represents the second barrier. The guard vessel containment is provided as an additional hermetic barrier in the extremely unlikely event that the integral core-unit itself would experience major failures. Without sources of pressure in the core-unit or in the containment itself, the containment is never challenged by pressure. Overheating of the containment is precluded by the balance of heat losses and heat generation even if the core-unit would fail. The containment itself is covered by thick horizontal steel radiation shield plates at the top. These plates also provide protection against extreme external events such as aircraft crash or explosion pressure waves, and provide an additional heat sink in any overheating scenario.

6. Instrumentation and Control Systems

The IMSR does not depend on electrical or even I&C systems for ultimate safety. For operability and investment protection, a few additional requirements do exist, such as freeze protection, as the fluoride salts have high freezing points. The normal plant operations systems provide this protection, in the form of pump trips, electrical (trace) heating, and thermocouple sensors. As with the steam turbine-generator, this type of equipment is available off-the-shelf from many suppliers. The pumps are the primary control units, as passive, flow driven control rods are utilized. This links the trip and protection logic to the pump rather than to a control rod drive and logic system. The result is a simple and easy to control system where the only real variable to control actively is the pump speed (trace heating is typically kept in automatic mode for normal operations). For example, to rapidly shut down the reactor, no control rod drives, drive power, hydraulic supply, or other support systems are needed; instead the pumps are simply tripped. The simplicity of this approach and the lack of dependence on any electrical systems also mean that cyber-security is easy to address.

As the IMSR does not depend on the electrical systems for ultimate safety, the design of a detailed plant electrical and control room system is eased. However, as with the steam turbine-generator, standard, off-the-shelf industrial equipment is expected to be used due to the generic nature of the requirements.

7. Plant Arrangement

The reactor auxiliary building and plant layout is shown in *Figure 4*. The IMSR core-unit is housed in a below grade silo. Surrounding the core-unit are the guard vessel cooling jacket, and at the top, a removable steel containment head plate. All these components are located below grade as well. Above the containment head, shield plates cover the core-unit for radiation shielding. An idle silo is provided next to the active silo, so that this silo can be loaded with a new core-unit while facilitating cool down of a spent core-unit. These two silos make up the main components of the reactor building. After draining the spent core-unit followed by a cool down period, the empty spent module is the lifted out and stored in long-term storage silos in an area adjacent to the reactor building. Inlets and outlets are provided for the (non-radioactive) coolant salt lines coming into and out of the core-unit, which transfer the heat to steam generators and re-heaters. The coolant salt lines are located inside the reactor building. The steam and reheat steam are utilized in a conventional, off-the-shelf turbine-generator system located in an adjacent building.

8. Plant Performance

The simple modular and replaceable core-unit should improve reliability and plant performance The core-unit has only a few moving parts (pumps) and these are redundantly fitted. Even premature replacement of a faulty module only mildly affects reliability and levelized cost. The use of an idling silo greatly increases plant capacity factor, as long cool down times (for radiation dose reduction) are possible without plant downtime.

The IMSR reactor building has a low profile and low mass, allowing rapid construction. The reactor building is a simple lightweight industrial building as it does not serve any major safety-related function - these are all provided by dedicated, robust, fail-safe systems.

Being a molten salt reactor (MSR), the IMSR has low fuel reload costs; fuel fabrication costs are zero, only extensive purification of the salts is required which, being bulk chemical processing, has a much lower cost than traditional high precision fuel fabrication with all its

quality and process control costs. Fuel recycling costs are also much lower with molten salt fuels, as costly fuel deconstruction and – reconstruction steps are avoided and also because simple, compact, low waste volume creating distillation and fluorination processes can be utilized. This makes it likely that spent IMSR fuel will be recycled.

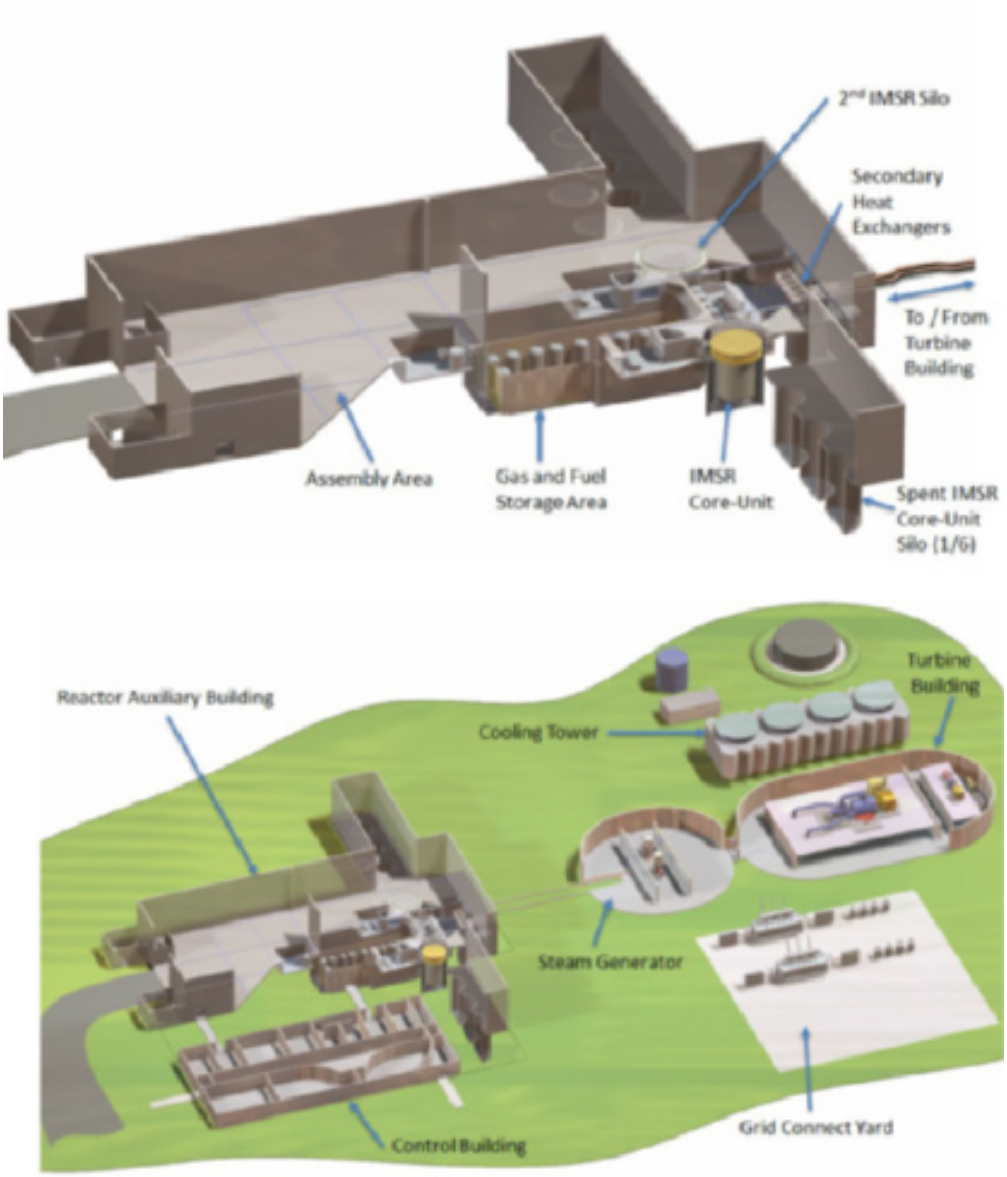


Figure 4: Cut-away view into the reactor auxiliary building and plant layout (Reproduced courtesy of Terrestrial Energy)

9. Design and Licensing Status

The current phase of work involves the support of a growing number of universities, third party laboratories and industrial partners. In addition, the Canadian nuclear regulator, the Canadian Nuclear Safety Commission, has started Phase 1 of a pre-licensing Vendor Design Review of the IMSR. This is an assessment of a nuclear power plant design based on a vendor's reactor technology.

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[1] For more information: www.terrestrialenergy.com

MSTW (Seaborg Technologies, Denmark)

1. Introduction

The Seaborg Technologies' Molten Salt Thermal Wasteburner (MSTW) is a thermal spectrum, single salt, molten salt reactor, operated on a combination of spent nuclear fuel (SNF) and thorium. It is envisioned to produce 100 MWe, or 115 MWe with a 2 stage turbine, from 270 MWth. The core outlet temperature is 700°C, but can go as high as 900°C for special uses, such as hydrogen production. The MSTW is designed around inherent safety features; no active measures are required to control the reactor under abnormal circumstances. The fully modularized MSTW is suitable for mass production. As a module reaches the end of its lifecycle, it will be extracted and returned for recycling in a central production facility after it has cooled down. The reactor core, including the graphite-based moderator, is projected to have a lifecycle of seven years, while the power plant will operate on the same batch of SNF for the 60 year facility lifetime. The MSTW is in the early design phase and Seaborg Technologies (ST) is focused primarily on neutronics, radiative transfer, computational fluid dynamics (CFD), and the physics of the design. A model of the underground reactor cave is shown in *Figure 1*.

Figure 1: Early model of the underground reactor cave (Reproduced courtesy of Seaborg Technologies).

2. Target Application

The MSTW is designed for electricity production, district heating/cooling, sea water desalination, etc. Due to the high outlet temperature it is well suited for synthetic fuel, as well as industrial process heat applications. Its high burnup and the fact that it is fuelled directly with spent nuclear fuel makes it a good option for spent nuclear fuel stockpile reduction. However, it can, without modification, operate on a wide array of different fuels.

3. Development Milestones

2014	Thermal wasteburner concept coined.
2015	Thermal wasteburner neutronic benchmark. Pre-conceptual design (whitepaper).

4. General Design Description

Design Philosophy

The MSTW was originally designed to address the three main concerns that resulted in the Danish ban on nuclear power in 1985, namely safety, waste, and proliferation. As the design has been developed and matured, the following three concepts together have been the essential philosophy around which the other features have been planned and implemented:

Safety: The system is placed in an optimised configuration, whereby any perturbation to such a system - from operator mistakes to major earthquakes - will move the system away from the optimum and thus result in an automatic power reduction and, if the situation is not alleviated, eventually reactor shutdown – see an illustration of one of these principles in *Figure 2*.

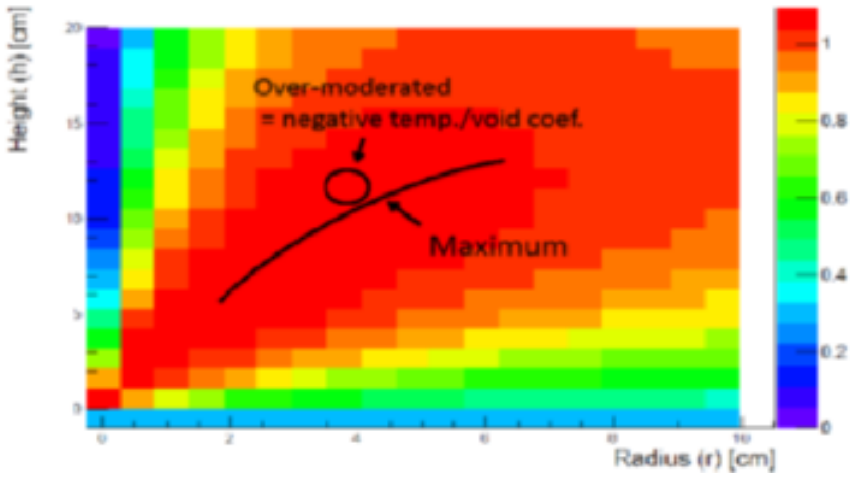

Figure 2: Optimal configuration principle and safety through physics – illustrated through a core geometry optimization of keff (Reproduced courtesy of Seaborg Technologies).

Waste: By combining SNF with thorium, the production of transuranic elements (TRU) is lower than the rate at which they are created. The net reduction is roughly five tonnes TRU over the 60 year plant lifetime.

Non-proliferation: To ensure high proliferation-resistance the MSTW is designed to be a single salt reactor wherein the chemical reprocessing system is incapable of extracting specific actinides. Furthermore, no actinide element will have an isotopic composition usable for nuclear weapons at any point in the fuel cycle.

Power Conversion Unit

The MSTW utilizes a coolant salt based on a core-integrated primary heat exchanger system. The coolant salt acts as heat storage and flows through the primary side of the steam generator, which delivers steam at 550°C to the single or two stage turbines (Rankine). The coolant salt and heat exchanger system also serves as core shielding and neutron reflector. The reactor core will passively shut down if overheated (negative thermal coefficient); therefore, loss of coolant accidents (LOCAs) will result in reactor automatically powering down. In case of a continued inability to cool the core, the fuel will eventually drain itself through a freeze plug to a passively cooled dump tank.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	Seaborg Technologies
Country of origin	Denmark
Reactor type	Molten Salt Reactor
Electrical capacity (MWe)	100 (1-stage turbine) or 115 (2-stage turbine)
Thermal capacity (MWth)	270
Expected capacity factor (%)	97.5%
Design life (years)	7 years component, 60 years plant life.
Plant footprint (m^2)	20000
Coolant/moderator	Graphite based
Primary circulation	Forced circulation
System pressure (MPa)	0.1 (sub-atmospheric)
Core inlet/exit temperatures (^oC)	600/700 – or 700/900 in the high temperature configuration
Main reactivity control mechanism	Negative temperature coefficient; graphite control rods
RPV height (m)	5
RPV diameter (m)	4
RPV or module weight (metric ton)	158 (including fuel and coolant)
Configuration of reactor coolant system	Integrated
Power conversion process	Rankine cycle
Process Heat Capabilities (High / Low T)	Different configurations possible (850°C /550°C)
Passive Safety Features:	Many: negative temperature coefficient; freeze plug to drain fuel to passively cooled dump tank; overflow system, etc.
Active Safety Features:	Redundant, but, SCRAM and borofluoride injection installed.
Fuel salt	Sodium-actinide fluoride (93% Th, 3.5% U 3.5% Pu)
Moderator height (m)	2.9
Number of moderator assemblies	91
Fuel enrichment (%)	Pre-processed SNF (U 1.1% fissile, Pu 69 % fissile)
Fuel burnup (GWd/ton)	250 (U and Pu - negligible Th cycle burning)
Fuel cycle (months)	720
Approach to engineered safety systems	Passive
Emergency Safety Systems	Inherent – passive
Residual Heat Removal System	Radiative (core), conduction (dump tank)
Refuelling outage (days)	Not Applicable (60 day module exchange, every 7 years)
Distinguishing features	Thermal spectrum wasteburner; Integrated heat-exchangers; Power production controlled by coolant pump speed
Modules per plant	1 or more
Target construction duration (months)	36
Design Status	Conceptual (mainly neutronics benchmarks done)

Reactor Core

The MSTW core is a graphite-based compound-moderator thermal reactor core as illustrated in *Figure 3*. The core is slightly over-moderated to ensure negative void and temperature reactivity coefficients. The graphite is coated with a metal and corrosion is reduced by

regularly circulating the fluoride salt through a fluoride burner and by adding a reducing anode to the fuel salt. The reactor operates at 12 kW per litre core, with a peak power density of 250 kW per litre fuel near the core centre. Swelling and corrosion of the graphite moderator is expected to be the limiting factor for the lifetime of the core module.

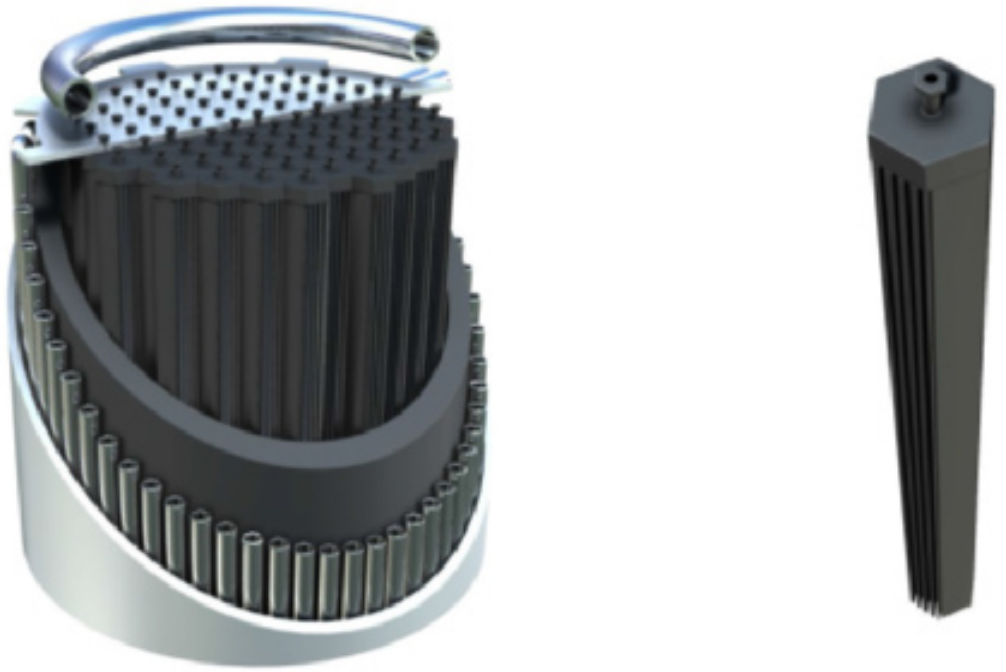

Figure 3: View of the inner reactor core and graphite moderator block (Reproduced courtesy of Seaborg Technologies).

Fuel Characteristics

The fuel salt is currently a eutectic sodium-actinide fluoride salt mixture, but sodium-rubidium-actinide or sodium-zirconium-actinide fluoride salts are being considered to reduce the actinide fraction in the salt, which will decrease the power density in the fuel salt and reduce pumping power requirements. Initially the actinide fuel mixture is 93% thorium, and 7% pre-processed SNF (3.5% uranium (1.1% enrichment) and 3.5% reactor-grade plutonium), but over time the plutonium "quality" decreases, and so only SNF is added and the thorium fraction decrease.

Reactivity Control

The strongly negative temperature coefficient ensures a saturation temperature level of the core, thus the power can be reduced by reducing cooling pump speed, or vice versa. The online refuelling slowly feeds fresh SNF into the salt, for continuous operation. Daily adjustment of saturation temperature level is achieved using a series of graphite fine-tuning rods, which can be inserted or extracted from the reactor. Using these rods, the reactor can be maintained critical at its operational temperature for up to two months, without any addition of fresh SNF.

A SCRAM system exists for the operators' convenience and for the 7-yearly core module replacement scheduled for each seven years of operation.

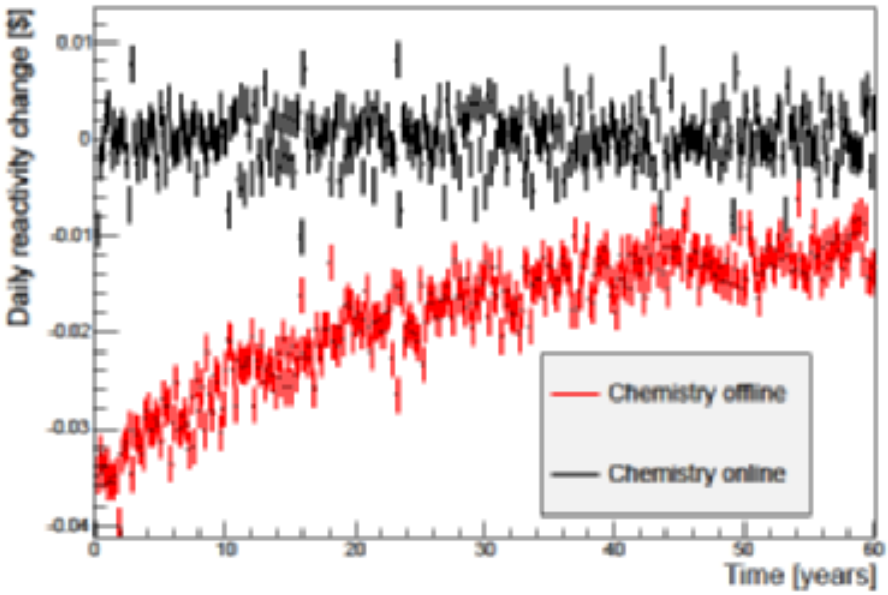

Figure 4: Simulation of reactivity control using the chemical system. New spent fuel is added while fission products are continuously extracted, handled and vitrified. (Reproduced courtesy of Seaborg Technologies)

Reactor Pressure Vessel and Internals

The reactor operates below atmospheric pressure above the fluid surface. However, the core, chemical unit, dump tank, and other critical components are planned to be contained in an airtight dome, able to withstand a significant pressure, both from internal and external events.

5. Safety Features

In the MSTW safety is guaranteed by physics rather than engineering. The reactor relies on natural convection, radiative heat transfer, and gravity for safety. The neutronics of the reactor is designed to be under optimal conditions. Then, any unexpected change to the system will result in a move away from the optimum, and thus a power reduction or shutdown of the reactor. All active safety systems have been made redundant.

Emergency core cooling system and decay heat removal system

In case of LOCA the core will heat up and the strong negative temperature coefficient will reduce reactivity and terminate power production without any operator interference. After shut down the core will heat up from residual heating, and passive cooling will be ensured through natural convection and radiative heat transfer for a substantial time. However, if main cooling is not restored, the reactor fuel will eventually drain itself (actively or passively) by melting a salt freeze plug. In the drain tank the fuel will solidify and is cooled by conduction to an underground heat sink.

Containment system

With the exception of noble gasses, noble metals, and cadmium, fission products are non-volatile in a fluoride salt. Note that this includes the hazardous caesium and iodine. The MSTW is designed so that critical temperatures are physically unreachable due to a number of reasons including the negative temperature coefficient, and the balancing of residual heating and radiative heat transfer. As such there are no scenarios where large amounts of non-volatile fission products can escape the fluid – even if the entire containment is breached.

The remaining volatile fission products are continuously extracted, handled and vitrified, and are therefore never present in significant amounts in the fluid. For this reason, the source term, even during extreme disasters, will be small and will pose a minimal off-site risk.

Despite the inherent safety, the reactor is designed with three barriers, where no irradiated component, coolant or fuel leaves the outer two barriers at any point.

6. Plant safety and Operational Performances

The neutronic optimum design philosophy provides excellent transient behaviour and outstanding inherent safety features, and the plant has outstanding load following capabilities. The heat storage in the coolant salt and the automatic saturation temperature of the reactor enables power production to be turned on and off as fast as the steam generators can be switched on and off. Furthermore, due to the degassing of noble gasses, and the relatively large fuel inventory, the reactor will not experience a poison-out during shutdown or power reduction.

7. Instrumentation and Control systems

The MSTW will have a series of control and monitoring features. Control will be based on a series of fine-tuning moderation rods, which will be placed near the edge of the reactor core so that an ejection event will reduce reactivity (loss of reflector). The reactivity span of these rods will be such that the reactor core saturation temperature can be adjusted within 200°C, and such that critical temperatures are not reachable through intentional or unintentional operator failures.

8. Plant Arrangement

Plant design is not yet finalized. However, all irradiated components, fuel, and coolant salt are contained in an approximately 8 m deep and 12 m wide underground area, covered with 2 m of concrete during operation. The turbine building, control room, and module handling area are placed above ground, above and around the underground area. The plant is envisioned to be operated by two operators, plus maintenance personnel for turbines and other plant components. During module change out, a team from the central manufacturing site will be dispatched to the site.

9. Design and Licensing Status

ST is currently focused on advanced multi-physics, neutronics, and early engineering details. The facility is planned as a green-field concept, and decommissioning considerations feature heavily in the design process, and most component parts are recyclable. Licensing is considered, and a plan has been drafted, however, at this early stage it is of lesser concern.

10. Plant Economics

The design is at a conceptual stage and the details of the plant economics are yet to be assessed. It is our belief that the redundancy of all active systems will result in significant cost reductions, through fewer engineering requirements and general simplicity. Also, the centralized manufacturing and recycling of modules is expected to benefit plant economics. Lastly, the short lifecycle of modules will heavily reduce requirements of materials and engineering. This is expected to significantly reduce the overnight cost, but will come at the expense of a larger over time cost. Postponing a large fraction of the plant cost from the overnight cost will be favourable from an economical point-of-view.

ThorCon (Martingale, International Consortium)

1. Introduction

ThorCon is a molten salt nuclear reactor [1]. Unlike all current operating reactors, the fuel is in liquid form. The molten salt can be circulated with a pump, and passively drained in the event of an accident. ThorCon operates at slightly above atmospheric pressure (similar to that in one's garden hose) and can thus use normal pipe thicknesses and easily automated, ship-style steel plate construction methods. The entire nuclear part of the plant is underground, as shown in *Figure 1*. This drawing shows a 1GWe ThorCon composed of four 250 MWe modules. The decay heat cooling towers are on the left. The underground nuclear island is center left. The yellow rectangles are hatches and are served by gantry cranes. The turbo-generator halls are center, and the switchyard is on the far right. The main cooling towers, if required, are to the right of the switchyard. The cranes allow periodic replacement of all critical components including the reactors and fuelsalt. The reactors and fuelsalt are transported by a special-purpose ship shown in the background.

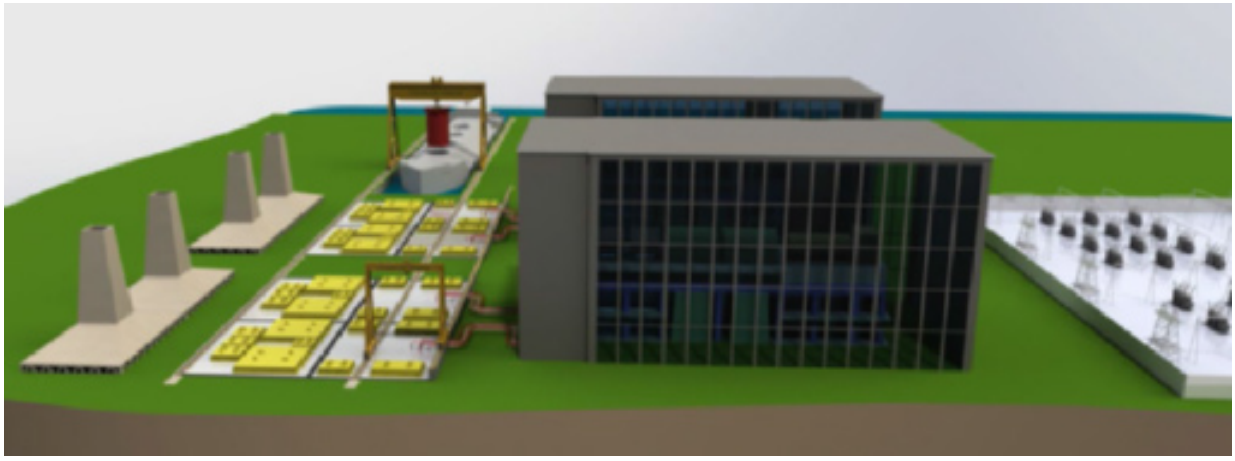

Figure 1: Birdseye View of 1 GWe ThorCon (Reproduced courtesy of ThorCon)

2. Target Application

The first planned application of ThorCon reactors is to produce electricity in developing nations. As such, the presumption is that the grid is still developing and the reactor must be capable of off-grid startup and thus must not depend on grid power. Cost is critical in these markets: ThorCon must be cost competitive with coal and be deployable at least as rapidly. A pre-feasibility study was initiated in Indonesia in 2016.

3. Development Milestones

2011	Conceptual design development
2016	Pre-feasibility study in Indonesia
2018	Pre-fission testing starts
2020	Fission testing starts
2022	Commercial operation starts

4. General Design Description

Design Philosophy

The following principles are followed in the ThorCon design [2,3]:

ThorCon is Walkaway Safe

ThorCon is a simple molten salt reactor with the fuel in liquid form. If the reactor overheats for whatever reason, it will automatically shut itself down and drain the fuel from the primary loop and passively remove the decay heat. There is no need for any operator intervention and the operators cannot prevent the draining and cooling. The reactor is 15 m underground. ThorCon has three gas tight barriers between the fuelsalt and the atmosphere. The reactor operates at slight over-pressure so that in the event of a primary loop rupture, there is no dispersal energy and also no phase change. The spilled fuel merely flows to a drain tank where it is passively cooled. The most troublesome fission products, including ^{90}Sr and ^{137}Cs, are chemically bound to the salt. They will end up in the drain tank as well.

ThorCon is Ready to Go

The ThorCon design should not need new technology development. ThorCon is a scale-up of the successful Molten Salt Reactor Experiment (MSRE). Currently the designers foresee no technical reason why a full-scale 250 MWe prototype cannot be operating within four years. The intention is to subject this prototype to all the failures and problems that the designers claim the plant can handle. As soon as the prototype passes these tests, commercial production can begin.

ThorCon is Rapidly Deployable

The entire ThorCon plant including the building is designed to be manufactured in blocks on a shipyard-like assembly line. These 150 to 500 ton, fully outfitted, pre-tested blocks are then barged to the site. A 1 GWe power station will require less than 200 blocks. Site work is limited to excavation and erecting the blocks. This should result in order of magnitude improvements in productivity, quality control, and build time. A single large reactor yard can turn out one hundred 1 GWe ThorCons per year. The philosophy is therefore that ThorCon is much more than a power plant; it is a new system for building power plants.

ThorCon is Fixable

The design does not foresee any complex repairs to be attempted on site. Except for the building everything else in the nuclear island is replaceable with little or no interruption in power output. Every four years the entire primary loop is changed out, returned to a centralized recycling facility, decontaminated, disassembled, inspected, and refurbished. The instrumentation design and monitoring system is designed to identify incipient problems before they can lead to failures. Major upgrades must be possible without significantly disrupting power generation. A nuclear power plant following such a change-out strategy can in principle operate indefinitely; but decommissioning should be little more than removing, but in this case not replacing, all the replaceable parts.

ThorCon is Cheaper than Coal

ThorCon requires far less resources than a coal plant. Assuming efficient, evidence-based regulation, ThorCon aims to produce clean, reliable, carbon-free electricity at less than the cost of coal.

Nuclear Steam Supply System

Figure 2 shows a cutaway view of the underground structure. ThorCon is divided into 250 MWe power modules. The drawing shows two such modules. Each module contains two

replaceable reactors in sealed Cans. The Cans are depicted in red in the drawing. They sit in silos. At any one time, just one of the Cans of each module is producing power. The other Can is in cooldown mode. Every four years the Can that has been cooling is removed and replaced with a new Can. The fuelsalt is transferred to the new Can, and the Can that has been operating goes into cool down mode.

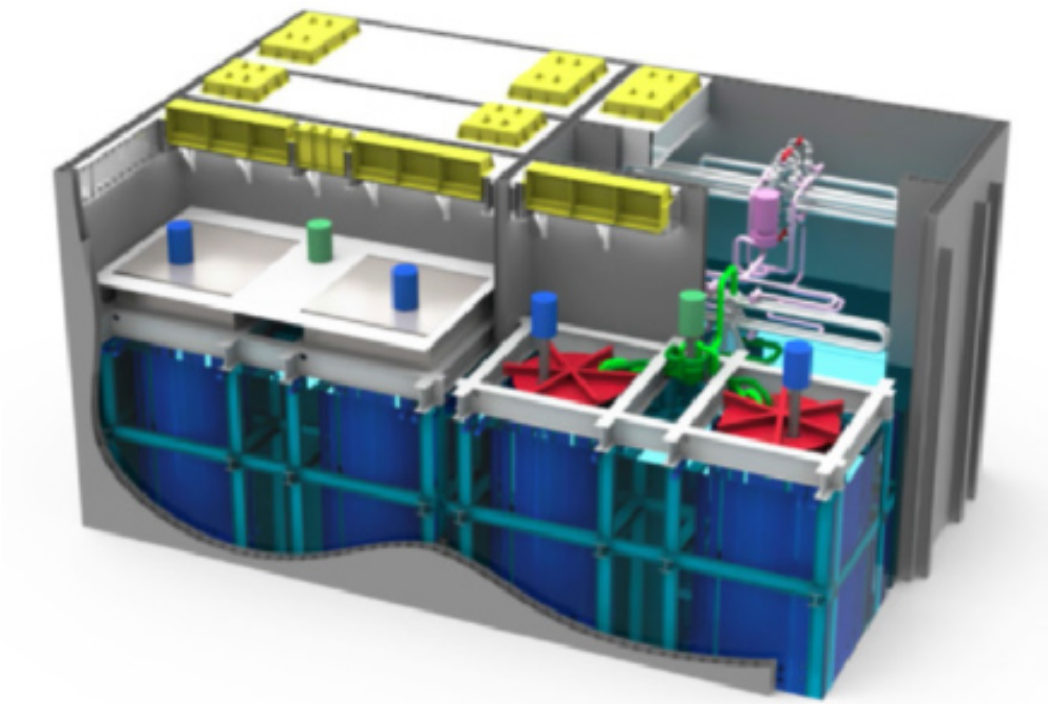

Figure 2: Cutaway view of two module silo hall (Reproduced courtesy of ThorCon)

Figure 3 takes a look inside a Can. The Can contains the reactor, which we call the Pot, a primary loop heat exchanger (PHX), and a primary loop pump (PLP). The pump (blue upper left) takes liquid fuelsalt — a mixture of sodium, beryllium, uranium and thorium fluorides — from the Pot (orange) at 704°C, and pushes the fuelsalt over to the PHX at a rate of just under 3000 kg/s (1 m^{3}/sec).

Flowing downward through the PHX (skinny blue), the fuelsalt transfers heat to a secondary salt, and is cooled to 564 ^{o}C in the process. The fuelsalt then flows over to the bottom of the Pot, and rises through the reactor core where the graphite moderator slows the neutrons produced by the fissile uranium, allowing a portion of the uranium in the fuelsalt to fission as it rises through the Pot, heating the salt and (indirectly) converting a portion of the thorium to fissile uranium.

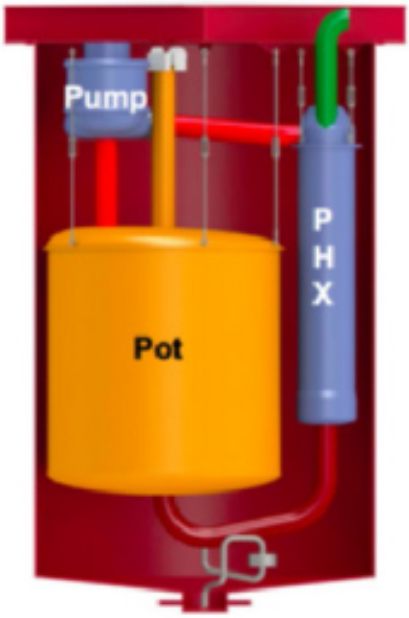

Figure 3: The ThorCon can: a pot, a pump, and a still (right)
(Reproduced courtesy of ThorCon)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	Martingale
Country of origin	International consortium planning first deployment in Indonesia
Reactor type	Thermal Molten Salt Reactor
Electrical capacity (MWe)	250 (per module)
Thermal capacity (MWth)	557 (per module)
Expected capacity factor (%)	> 90%
Design life (years)	80 years
Plant footprint (m^2)	20000 for 500 MWe
Coolant/moderator	NaF, BeF2 salt, graphite moderated
Primary circulation	Forced circulation
System pressure (MPa)	0.3 at primary loop max stress point, 1.05 at exit of primary
Core inlet/exit temperatures (^oC)	565 / 704
Main reactivity control mechanism	Negative temperature coeff; salt flow rate, control rod insertion
RPV height (m)/ diameter (m)	12 (includes full primary loop and off-gas) / 8
RPV or module weight (metric ton)	400
Configuration of reactor coolant system	Four loops: Fuel salt, secondary salt, solar salt, steam.
Power conversion process	Rankine steam
Passive Safety Features:	Fully passive shutdown and cooling. 72 day grace period.
Active Safety Features:	Drain fuel salt, shutdown rods.
Fuel salt	12% heavy metal in NaBe salt.
Heavy metal composition	80% Th, 16% U-238, 4% U-235
Makeup salt	12% uranium (enriched to 19.7%) in NaBe salt
Fuel enrichment (%)	19.7
Fuel burnup (GWd/ton)	256 GWd/ton U
Fuel cycle (months)	96
Approach to engineered safety systems	Avoid them. Physical limit on fuel addition rate; H/W limit on pump speed change rate.
Number of safety trains	Three means to remove decay heat. Two are fully passive.
Emergency Safety Systems	Three levels of containment, 3 cooling systems, 2 shutdown
Residual Heat Removal System	primary cooling to ocean; natural circulation to air; steam release
Refueling outage (days)	approximately 7
Distinguishing features	Low cost, full passive safety, short construction time
Modules per plant	1-4 per building, arbitrary per generating station
Target construction duration (months)	6
Seismic design	Target 0.8 pga
Design Status	Finishing conceptual design

The Pot pressure is 3 bar gage at the maximum stress point. The outlet temperature of 704 ^{o}C results in an overall plant efficiency of about 45%, and a net electrical output per Can of 250MW. The Can's net consumption of fissile uranium is 112 kg per year. The Can (red) is a cylinder 11.6 m high and 7.3 m in diameter. It weighs about 400 tons. The Can has only one major moving part, the pump impeller.

Directly below the Can is the Fuelsalt Drain Tank (FDT) (green). In the bottom of the Can is a fuse valve (grey). The fuse valve is merely a low point in a drain line. At normal operating temperatures, the fuelsalt in the fuse valve is frozen creating a plug. But if the Can heats up for any reason, the plug will thaw, and the fuel salt will drain to the FDT. Since the drain tank has no moderator, fission will stop almost immediately. This drain is totally passive. There is nothing an operator can do to prevent it.

A critically important feature of ThorCon is the silo cooling wall (blue). The silo wall is made up of two concentric steel cylinders, shown in blue in *Figure 4*. The annulus between these two cylinders is filled with water. The top of this annulus is connected to a condenser in a decay heat pond. The outlet of this condenser is connected to the basement in which the Can silos are located. This basement is flooded. Openings in the bottom of the outer silo wall allow the basement water into the bottom of the annulus. The Can is cooled by thermal radiation to the silo cooling wall. This heat converts a portion of the water in the wall annulus to steam. This steam/water mixture rises by natural circulation to the cooling pond, where the steam is condensed, and returned to the bottom of the cooling wall via the basement. In this process, some of the water in the pond is evaporated. The decay heat cooling towers return almost all this water to the pond.

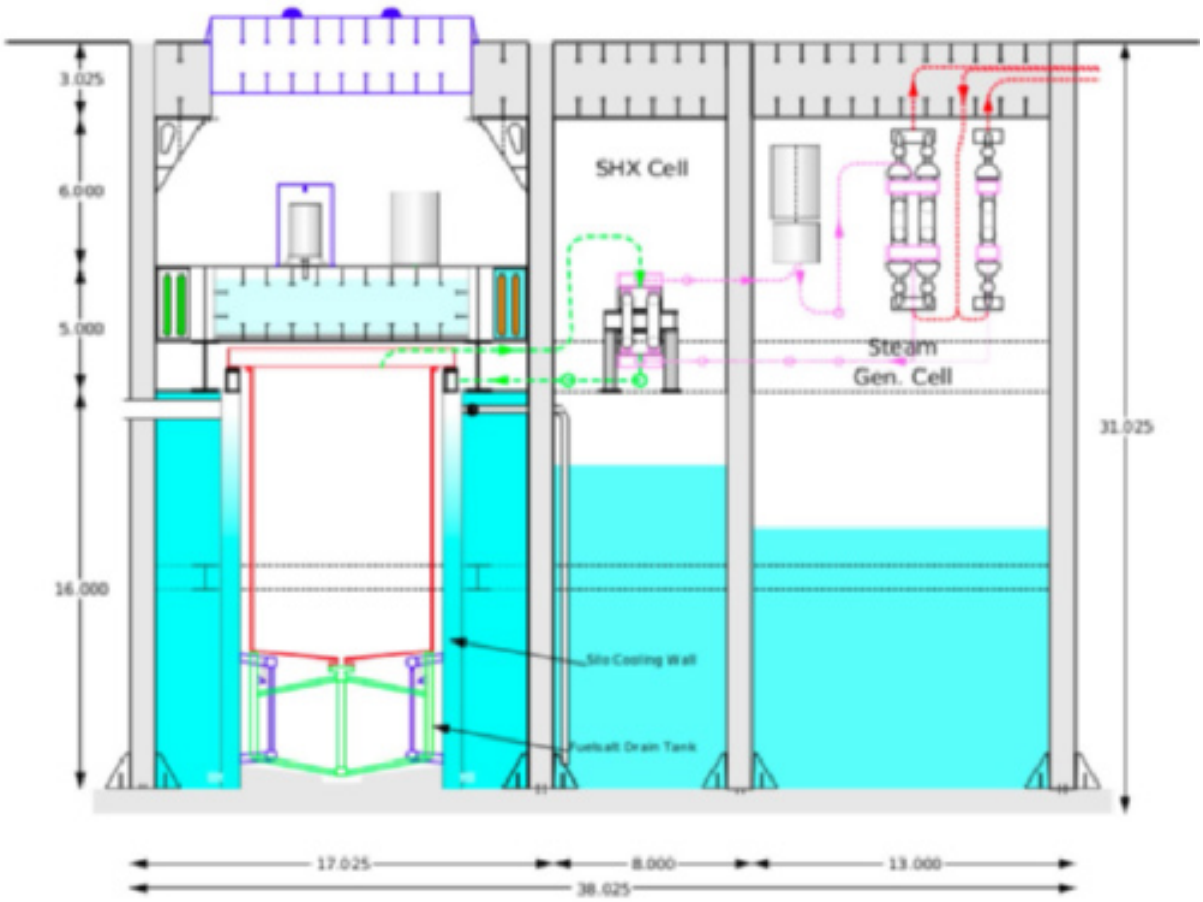

Figure 4: Silo Hall Cross-Section (Reproduced courtesy of ThorCon)

The silo cooling wall also cools the Fuelsalt Drain Tank (FDT). The drain tank is tall, thin rectangular trough that has been wrapped into a circle. This arrangement provides sufficient radiating area to keep the peak tank temperature after a drain within the limits of the tank material. This cooling process is totally passive, requiring neither operator intervention nor any outside power.

Each Can is located in a Silo. The top of the Silo is 14 m underground. *Figure 4* shows the secondary salt loop in green. The secondary salt is a mixture of sodium and beryllium fluoride containing no uranium or thorium. Hot secondary salt is pumped out of the top of the Primary Heat Exchanger to a Secondary Heat Exchanger where it transfers its heat to a mixture of sodium and potassium nitrate commonly called solar salt from its use as an energy storage medium in solar plants. The solar salt, shown in purple in *Figure 4*, in turn transfers its heat to a supercritical steam loop, shown in red.

ThorCon is a high temperature reactor that translates to thermal efficiency of up to 45% compared to about 32% for standard light water reactors. This reduces capital costs and cuts cooling water requirements by 60%. It also allows us to use the same steam cycle as a modern coal plant.

Reactor Core
The reactor core is inside the pot as shown in *Figure 3*. The core is 90% filled with graphite slabs, the moderator. The core is 5m diameter, and 5.7m high.

Fuel Characteristics
The fuelsalt is NaF-BeF2-ThF4-UF4 76/12/9.5/2.5 where the uranium is 19.7% enriched. As fissile is consumed more fissile (either U233 or Pu239) is generated but not enough to replace the fuel burned. The reactor has no excess reactivity, no burnable poisons, and no poison control rods. Makeup fuel must be added daily.

Fuel Handling System
Makeup fuel is added by applying gas pressure to the makeup fuelsalt tank which forces makeup fuel into the primary loop. The makeup fuel composition is NaF-BeF2-UF4 76/12/12 where the uranium is 19.7% enriched. The makeup fuel addition rate is physically limited to ensure the adding reactivity rate stays within acceptable limits. Excess fuelsalt flows into a holding tank.

Reactivity Control
The primary reactivity control is temperature and fuelsalt flow rate. For slow reactivity control makeup fuelsalt (or makeup fertile salt with no fissile) additions allow modest daily increase or decrease of the reactivity.

Reactor Pressure Vessel
The ThorCon reactor is never under high pressure so that the typical term Reactor Pressure Vessel does not really apply. In the design the Can plus the Fuel Drain Tank fulfil the same function since all radioactive material (except tritium) should be contained within these structures. Since no high pressure is present that can act as a driving force to spread the content into the environment, the RPV does not have the central safety importance in an MSR that it does in a LWR.

5. Safety Features

The ThorCon design combines a negative temperature coefficient with a large margin between the operating temperature of 700 ^{o}C and the fuelsalt's boiling temperature (1430 ^{o}C) to assure passive, totally unavoidable, shutdown and cooling. In any event that raises the temperature of the salt much above the operating level, the reactor will automatically shut itself down. If the high temperature persists, the fuse valve will thaw and drain the fuel from the primary loop to the drain tank, where the silo cooling wall will passively remove the decay heat.

There is no requirement for operator (or control system) intervention at any time since there are no valves to realign, pumps to activate, or any other actions to be taken. In fact there is nothing the operators can do to prevent the drain and cooling. The decay heat is transferred to an external pond which has sufficient water for 72 days cooling. After 72 days without any intervention the water in the pond will be running low. Adding more water is simple since the pond is accessible and at atmospheric pressure. If the pond cooling line is lost, there is enough water in the basement to handle the first 30 days of decay heat

Release Resistant The ThorCon reactor is 15 m underground. ThorCon has three gas tight barriers between the fuelsalt and the atmosphere. Two of those barriers are more than 10 m underground. ThorCon reactor operates at near-ambient pressure. In the event of a primary loop rupture, there is no dispersal energy and no phase change and no vigorous chemical reactions (like zirconium and steam). The spilled fuel merely flows to the drain tank where it is passively cooled. Moreover, the most troublesome fission products, including iodine-131, strontium-90 and cesium-137, are chemically bound to the salt. They will end up in the drain tank as well. Even if all three barriers are somehow breached, almost all these salt seekers will not disperse.

No separate, spent fuel storage ThorCon uses an eight-year fuelsalt processing cycle, after which the used salt is allowed to cool down in the non-operating Can for four years, eliminating the need for a separate, vulnerable spent fuel storage facility. The fuelsalt that is cooling is as well-protected as the fuelsalt that is currently being burned.

Four loop separation of steam and fuelsalt. ThorCon employs four loops in transferring heat from the reactor to the steam turbine. The solar salt loop captures any tritium that has made it to the secondary loop, and more importantly ensures that a rupture in the steam generator creates no harmful chemicals and harmlessly vents to the Steam Generating Cell via an open standpipe.

6. Plant safety and Operational Performances

Load following is accomplished by changing the pump speed while keeping the temperatures relatively constant. The requirements for load following are drawn from EU Utility Requirements for Light Water Reactors (5% full power per minute from 50% to 100% full power). Since the off-gases are continuously removed xenon poisoning and oscillations are not an issue. No neutron poisons are used in the control of the reactor which reduces fuel consumption.

7. Instrumentation and Control systems

The instrumentation and control and even the operators are not safety critical in the design. ThorCon instrumentation uses a multiplicity of sensors to record and report the condition of the power plant. Statistical process control is used to track trend lines and detect incipient failures such as bearing wear. A common control center for all modules in a power plant

minimizes staffing requirements. Continuous reporting of plant conditions to a central engineering facility monitoring conditions at all plants allows for fleet wide analysis and makes expert advice available to any plant experiencing unusual conditions as well as monitoring plants for unusual activity.

8. Plant Arrangement

Sections 1 and 4 and *Figures 1-4* describe the plant arrangement. The control building is shared among all modules in a power plant. In the most common deployment envisaged two power modules will drive a single 600 MWe turbine/generator. This size was chosen as it should be competitive priced per MWe while also suitable for smaller deployments.

9. Design and Licensing Status

Conceptual design is nearly complete. License discussions have been started with Bapeten (Indonesian regulator).

10. Plant Economics

In order to be successful, ThorCon must be able to compete with coal with zero CO_2 price. A first comparison was therefore made with the cost of coal plants. Since ThorCon uses the same turbine island and electrical side as a standard 500 MWe supercritical coal plant, the cost comparison was made on the steam generation parts. When the steel and concrete requirements of a 1 GWe plant was compared, the ThorCon nuclear island requires one-fifth as much steel and one-third as much concrete as the coal plant boiler, fuel handling, and ash handling systems.

Moreover, almost all the ThorCon concrete is non-structural, and simple construction of concrete dumped into the steel sandwich walls; while a large part of the coal plant concrete is slow, labour intensive, reinforced concrete. Based only on this resource basis the overnight cost of the ThorCon nuclear island should be less than one-third that of the coal plant's steam generation side.

Other costs also need to be taken into account. A 1GW ThorCon requires 1500 tons of very high quality graphite, 1300 tons of SUS 316, 220 tons of the superalloy Haynes 230, 2,500 tons of lead, and 200000 m^3 of excavation. But these and other adjustments add a little more than 100 million dollars to the cost of a 1 GWe plant or about $100/kW. Overall, the resource cost of the ThorCon nuclear island is less than one-half that of the coal plant's steam generation systems, or less than a $500 per kW. Based on this assessment ThorCon is cheaper than coal, both in capital costs and in fuel costs.

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[1] Complete ThorCon website: http://thorconpower.com
[2] Succinct description of philosophy and design: http://thorconpower.com/docs/domsr.pdf
[3] Designers' essays on molten salt reactor aspects: http://thorconpower.com/library/documents

FUJI (International Thorium Molten-Salt Forum, Japan)

1. Introduction

The Molten Salt Reactor (MSR) uses molten salt, in general molten fluoride salt, as liquid fuel and coolant. MSR was originally developed at Oak Ridge National Laboratory (ORNL) in 1960s, and three experimental MSRs were constructed. One of them was operated for 4 years without severe problems. Thus, it is verified that the MSR technology is feasible.

MSR-FUJI was developed since the 1980s by a Japanese group (now, International Thorium Molten-Salt Forum: ITMSF), based on the ORNL's results with the view to deploy it widely in the world [1,2].

Molten salt is a liquid, which is in general a melted chemical compound of acid and alkali at high temperature. Molten salt is stable and inert at high temperature, and can be used at very low pressure. Since core meltdown or steam/hydrogen explosion is impossible, high safety can be achieved.

MSR-FUJI is size-flexible as from 100 MWe to 1000 MWe. But, a latest and typical design (FUJI-U3) is 200 MWe, which can be categorized as small-sized reactors with modular designs (SMR). The thermal output of FUJI-U3 is 450 MWt and thus a 44% thermal efficiency can be attained. In addition to the expected high efficiency the simple core structure and high fuel efficiency should facilitate a favourable economic performance.

A schematic diagram of the MSR-FUJI is shown in *Figure 1*.

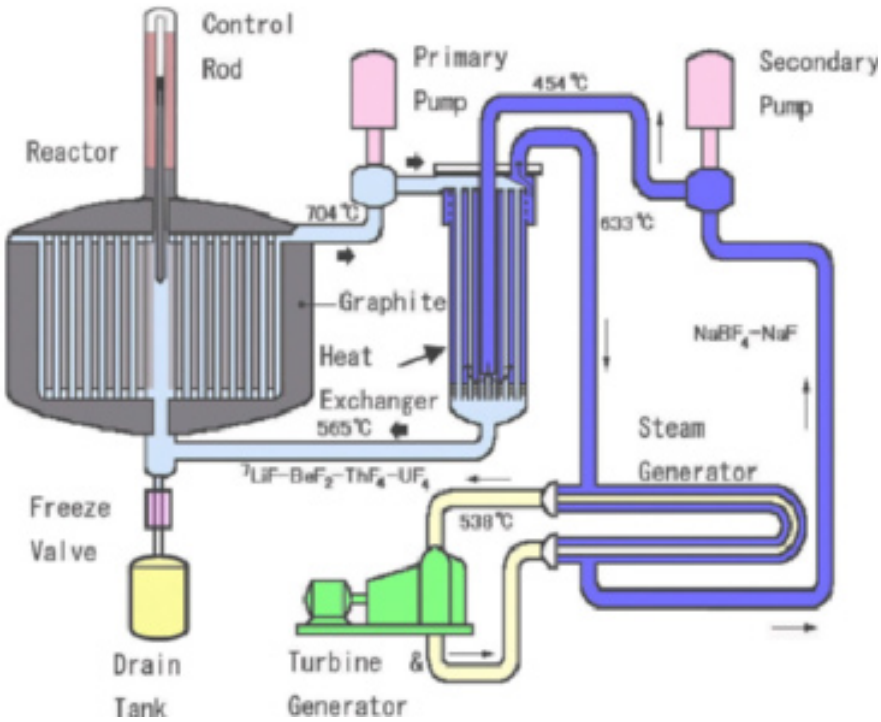

Figure 1: Schematic diagram of MSR-FUJI (Reproduced courtesy of ITMSF)

Molten fuel salt can contain thorium (Th) as fertile material and ^{233}U as fissile material, and based on this the FUJI-U3 design can attain a self-sustaining fuel cycle with a conversion factor of 1.0. Since MSR-FUJI applies the Th-cycle, generation of plutonium (Pu) and minor actinide (MA) is very small compared with Light Water Reactors (LWR). Furthermore, it can consume Pu as the fissile material, and can thus contribute to reduce the proliferation risk caused by Pu from LWR spent fuel. It can also be used to transmute long-lived MA to shorter ones.

2. Target Application

MSR-FUJI can be applied not only to electricity generation, but also to transmutation of Pu and/or MA. Besides these purposes, it can be used as a heat source for water supply by desalination of seawater or for hydrogen production, utilizing its high exit temperature of 704 °C.

3. Development Milestones

1980's	Conceptual designs of MSR-FUJI have been started
1980's	Accelerator Molten-Salt Breeder (AMSB) design for a large production of fissile material (similar to Accelerator Driven System ADS)
Until 2008	Several designs, such as a pilot plant (mini-FUJI), a large-sized plant (super-FUJI), a Pu-fueled plant (FUJI-Pu),
Recent	The latest SMR plant (FUJI-U3), The MSR-FUJI as in this text refers to the 200 MWe FUJI-U3 design

4. General Design Description

Design Philosophy

The design philosophy of MSR-FUJI is to achieve a high level of safety, good economic performance, contributing to non-proliferation, and to achieve fuel cycle flexibility.

MSR-FUJI is based on the ORNL's results, and has been optimized as a small sized plant and further simplified by removing the online reprocessing facility. Based on the operating experience at three experimental MSRs in ORNL, it has been verified that MSR-FUJI is feasible. The steam generator (SG) is however a major unverified component but it can be developed based on Fast Breeder Reactor (FBR) experience and the recent supercritical power station technology.

MSR-FUJI adopts a passive safety system to improve the safety, reliability as well as the economics. Molten fuel salt can be drained to a sub-critical drain tank through a freeze valve. Since gaseous fission products (FP) are always removed from molten fuel salt, the risk at accidents is minimized. MSR-FUJI is operated at very low pressure (0.5 MPa), and a thick reactor vessel and pipes are not required. There are no fuel assemblies or complex core internal structure with the only component within a reactor vessel being the graphite moderator. Based on these design principles in-factory fabrication would be simple.

Nuclear Steam Supply System

The nuclear steam supply system (NSSS) consists of a reactor core, pipes, pumps, a heat exchanger (HX), and a steam generator (SG), which supplies steam to a turbine/generator (T/G), as is shown in *Figure 1*. This figure shows only one loop, but a loop can be redundant depending on a plant size or a need for flexibility.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	International Thorium Molten-Salt Forum: ITMSF
Country of origin	Japan
Reactor type	Molten Salt Reactor
Electrical capacity (MWe)	200
Thermal capacity (MWth)	450
Expected capacity factor (%)	75% for daily load-following operation
Design life (years)	30
Plant footprint (m^2)	<5000 (R.B. + SG.B. + TG.B.)
Coolant/moderator	Molten fluoride / Graphite
Primary circulation	Forced circulation by pump
System pressure (MPa)	0.5 (by pump head)
Core inlet/exit temperatures (^oC)	565 / 704
Main reactivity control mechanism	Control rod, or pump speed, or fuel concentration
Reactor Vessel height (m)	5.4 (inner)
RV diameter (m)	5.34 (inner)
RV weight (metric ton)	60 (made of Hastelloy N)
Configuration of reactor coolant system	Primary loop and secondary loop
Power conversion process	Rankine Cycle
Cogeneration / Process Heat Capabilities (High / Low T)	Possible, utilizing high exit temperature
Passive Safety Features:	Freeze valve to drain tank.
Active Safety Features:	Control rod scram. ECCS is not required
Fuel type/assembly array	Molten salt with thorium and uranium
Fuel assembly active length (m)	No fuel assembly
Number of fuel assemblies	No fuel assembly
Fuel enrichment (%)	2.0 equivalent (0.24%^{233}U + 12.0%Th). Pu or LEU can be used.
Fuel burnup (GWd/ton)	No mechanical limitation for burnup
Fuel cycle (months)	Continuous operation is possible
Approach to engineered safety systems	Passive safety
Number of safety trains	ECCS/CCS/ADS are not required
Emergency Safety Systems	Ditto
Residual Heat Removal System	Passive cooling without electricity (required at drain tank)
Refueling outage (days)	<30 (Refueling shutdown is not required)
Distinguishing features	High safety, high economic performance, contribution to non-proliferation, and fuel cycle flexibility
Modules per plant	(Not decided)
Target construction duration (months)	(Not decided)
Seismic design	Same as LWRs
Core damage frequency (per reactor-year)	Core meltdown is impossible
Design Status	3 experimental MSRs were built. Detailed design has not started

MSR-FUJI is designed to produce an exit temperature of 704°C in molten fuel salt, and its heat is transferred to the secondary salt through a HX. Then, its heat produces 538°C supercritical steam at a SG, and generates electricity by a supercritical T/G. Owing to its high temperature, MSR-FUJI can achieve 44% thermal efficiency.

Since molten salt is used as fuel and coolant, loss of coolant accident (LOCA) may need a new definition for MSR. In case of pipe break, leaked molten salt is drained to an emergency drain tank without passing through a freeze valve. Even in this case, a molten salt loop returns to atmospheric pressure when a pump stops, and a pressurization accident is incredible, owing to its low vapor pressure. Therefore, an emergency core cooling system (ECCS), containment cooling system (CCS), makeup water pools, and automatic depressurization system (ADS) are not required.

The primary loop (molten fuel salt loop) is operated with forced circulation by a centrifugal pump during normal operation. The system also has a natural circulation capability in emergency conditions.

Reactor Core

A reactor vessel is cylindrical in shape. The core structure is made up by hexagonal shaped graphite moderator blocks. The blocks contain holes that serve as the flow paths of the molten fuel salt that flow upwards through the blocks circulated by the primary pump. The molten fuel salt then goes to a heat exchanger to transfer the heat to the secondary coolant salt.

The concentration of the fuel composition can be adjusted at any time through the fuel concentration adjustment system. Since there are no fuel assemblies in the core, refueling shutdown is not required, and continuous operation is possible. In order to achieve a core conversion factor of 1.0 it is recommended to refresh the fuel salt every 7 years. Periodic maintenance shutdown will of course be required as in any power plant.

Fuel Characteristics

The molten fuel salt is a liquid form of fluoride ($LiF\text{-}BeF_2$) with ThF_4 and a small amount of $^{233}UF_4$. A typical composition is $LiF\text{-}BeF_2\text{-}ThF_4\text{-}^{233}UF_4$ (71.76-16-12-0.24 mol%).

Molten fluoride can be used at very low pressure owing to its very high boiling temperature and very low vapor pressure. The melting temperature of the above fuel composition is 499°C. It can dissolve uranium (U) or Pu as fissile material so that low enriched uranium (LEU) or Pu can be used. Owing to unique features of molten fuel salt, fuel assembly fabrication is not required, and radiation damage or fuel cladding failure does not occur.

Reactivity Control

Reactivity control for long-time operation can be performed anytime by a fuel concentration adjustment system. In normal daily operation, reactivity or power level can be controlled by core flow or by core temperature. Control rods are withdrawn in normal operation and are inserted by gravity in case of emergency shutdown.

Reactor Vessel and Internals

The reactor vessel is made of Hastelloy N. Since the operating pressure is very low (0.5 MPa) a “pressure” vessel is not required and the reactor vessel wall thickness is about 5 cm. Only one component, the graphite moderator blocks, is present within the core internal region.

Secondary salt loop
The secondary loop adopts molten salt of $NaBF_4$-NaF. This secondary loop is circulated by a centrifugal pump and removes heat from the primary loop through a heat exchanger to the steam generator. Since the pressure of both the primary and secondary loop is very low, the danger of rupture is minimized. In case of a pipe break molten salt is drained to a drain tank.

Steam Generator and Turbine Generator
A steam generator (SG) of MSR-FUJI adopts a shell and tube design. A U-shaped shell contains a secondary salt flow, and steam flows inside of multiple tubes within a shell. The secondary salt loop at 633°C provides heat to the SG that generates 252 kg/cm^2 steam at 538°C fed to the supercritical turbine generator (T/G) to produces 200 MW electricity.

5. Safety Features

The MSR-FUJI design has very favourable safety characteristics that essentially exclude the possibility of severe accidents based on the following unique features:

- The primary and secondary loops are operated at a very low pressure, which essentially eliminates accidents such as system rupture due to high pressure.
- The molten salt is chemically inert, that is, it does not react violently with air or water, and is also not flammable. The corrosion of Hastelloy N can be minimized by the appropriate chemical control and maintenance of the molten salt.
- Pressure increase in a primary loop is incredible, because a boiling temperature of the fuel salt is very high (about 1,400°C) compared with the operating temperature (about 700°C).
- Since there is no water within the containment, there is no possibility of high pressure by steam generation, and therefore no possibility of hydrogen explosions at any accidental conditions.
- The fuel salt is drained to a drain tank through a freeze valve, if required. In case of a rupture in the primary loop, the spilled fuel salt is drained to an emergency drain tank without passing a freeze valve.
- The fuel salt only reaches criticality within the graphite core (with the appropriate moderation). In an accident the fuel salt is drained to a drain tank designed so that a re-criticality accident does not occur.
- MSR has a large negative reactivity coefficient of a fuel salt temperature that can suppress an abnormal change of the reactor power. Although a temperature coefficient of graphite is positive, it does not affect the safety, because the heat capacity of graphite is large enough.
- Since gaseous fission products (FP) can be removed by separation from the fuel salt, the danger due to the release of radioactivity from the core at accidental conditions can be minimized.
- Since fuel composition can be adjusted when necessary, the excess reactivity and the reactivity margin to be compensated by control rods are small. Therefore, reactivity requirements for control rods are also small.
- The delayed neutron fraction of ^{233}U is lower than that of ^{235}U and some of delayed neutrons are generated outside the core. However, safe control of the reactor is possible because of a large negative reactivity coefficient and limited potential reactivity insertion.
- Since there is no airflow and no heat source within the core when the fuel salt is drained at accidental conditions, self-sustained graphite oxidation does not occur.

Engineered Safety System Approach and Configuration
Many of the safety features were described above. Therefore, redundant and diverse emergency core cooling systems (ECCS), makeup water pools, and automatic depressurization system (ADS) are not required. In order to protect against a freeze accident in a molten fuel salt loop, a high temperature containment is equipped.

Decay heat removal system
In normal shutdown condition, decay heat is transferred to a secondary loop and a steam-line loop, and disposed to the ultimate heat sink (seawater for example). If all pumps in a primary or secondary loop stop, fuel salt is drained to a drain tank through a freeze valve. Decay heat at the drain tank is cooled by a passive heat removal system, and finally its heat is disposed to the outside environment through an air-cooled system that does not require electricity.

Emergency core cooling system
As is explained above, redundant and diverse emergency core cooling systems (ECCS) and makeup water pools are not required. This would simplify the plant, and eliminate concerns of failures in safety systems.

Containment system
Since the risk of pressurization accidents is incredible, the containment size can be minimized. Although molten salt is not flammable, inert gas (N_2) is enclosed within a containment in order to maintain fuel salt purity in case of a pipe break accident. The MSR-FUJI design has three levels of containment. The first is the reactor vessel and pipes made of Hastelloy N. The second is a high temperature containment composed of three layers, which contains a reactor vessel, pipes, and a heat exchanger, as is shown in *Figure 2*. In order to avoid a freezing accident, this containment is equipped with heaters. The third level is a reactor building composed of two layers. As explained above, a pressurization accident is incredible due the low vapor pressure. Therefore, a containment cooling system (CCS) and makeup water pools are not required.

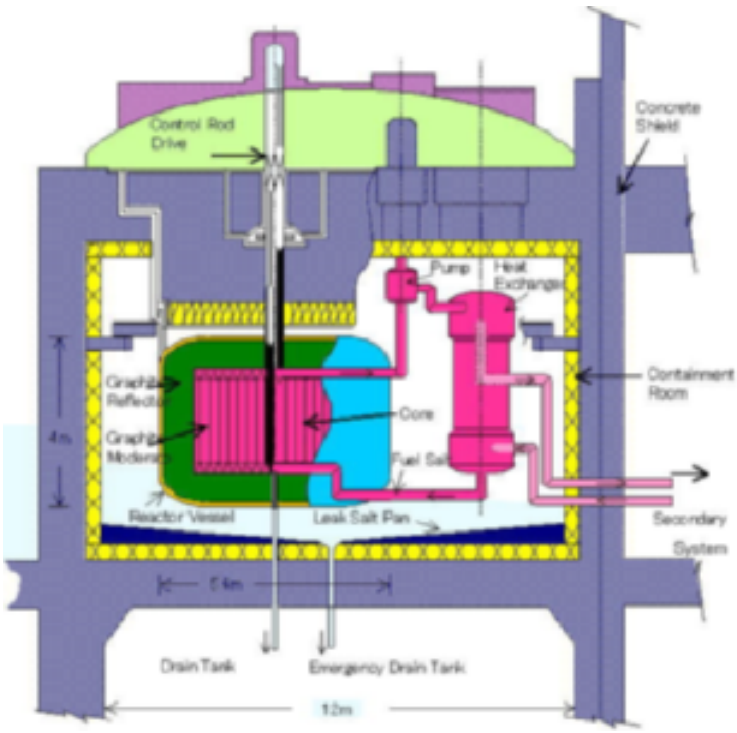

Figure 2: Vertical cross section of primary system of MSR-FUJI (Reproduced courtesy of ITMSF).

6. Plant safety and Operational Performances

Overall safety is described above. In case of a station blackout (SBO: Loss of all AC electricity) the MSR-FUJI can be shut down and cooled without electricity. Core meltdown or steam/hydrogen explosion is physically excluded by design and no ECCS is needed. As for long-time operation, reactivity can be controlled anytime by a fuel concentration adjustment system.

In normal daily operation, the power level can be controlled by core flow or by core temperature control. That is, load following is easily performed without using control rods. Control rods are withdrawn in normal operation, and are inserted by gravity in case of emergency shutdown.

7. Instrumentation and Control systems

Instrumentation and control (I&C) systems in the MSR-FUJI design are the same as for recent LWR designs. It must support operators in making decisions and efficiently operating the plant during plant start-up, shutdown, normal operation, surveillance testing, and accidental situations. It adopts the man-machine interface more useful, and expands the scope of automatic control.

I&C systems include an alarm system, an information processing system, a reactor protection system, an engineered safety equipment control system, power and process control systems, sensors and instrumentation, an radiation monitoring system, and so on.

8. Plant Arrangement

Major buildings of MSR-FUJI are a reactor building, a steam generator building with a main control room, and a turbine-generator building as shown in *Figure 3*. An auxiliary building is not shown.

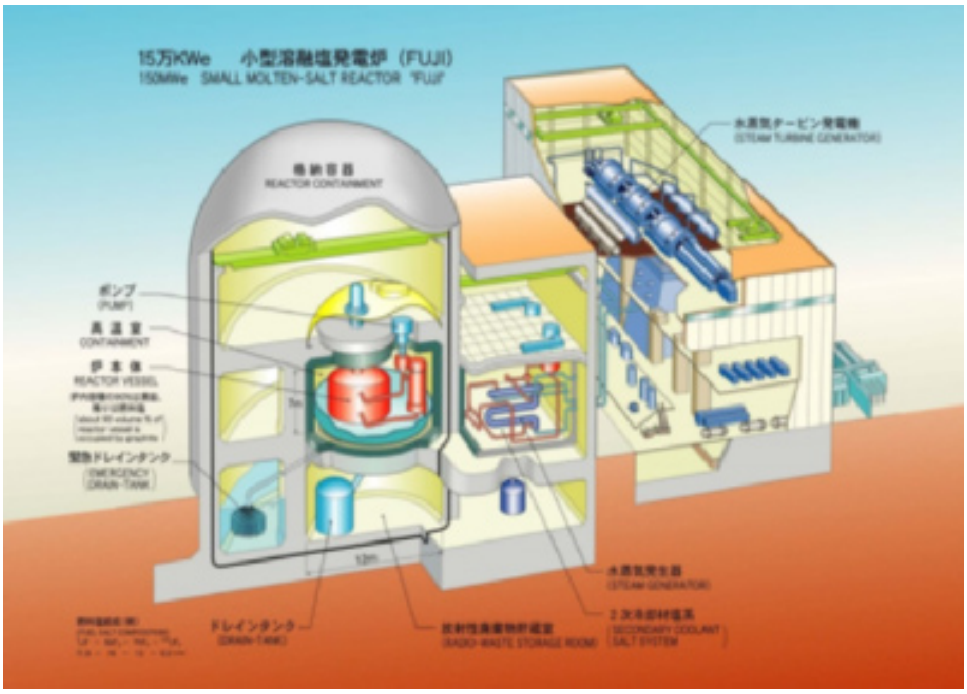

Figure 3: Bird-eye view of MSR-FUJI (Reproduced courtesy of ITMSF).

Reactor building

The reactor building contains a high temperature containment, drain tanks, a radio-waste storage, and other facilities required for the reactor. This reactor building is a cylindrical

shape with a hemispherical dome, which is made of concrete with steel liner as its inner layer. The reactor building is founded on a common base-mat together with other buildings.

Control building
The main control room (MCR) is located at a steam generator building, which is next to a reactor building. The MCR is a key facility to cope with normal and emergency situations, so it is designed to ensure that plant personnel successfully perform the tasks according to the proper procedures.

Balance of plant:

Turbine Generator Building
The turbine generator (T/G) building contains the supercritical turbine and generator, which produce electricity. Also, it contains condensers for disposed steam. The condensers use outside water (for example, sea water) for cooling.

Electric Power Systems
These systems include the main generator, transformers, emergency diesel generators (EDG), and batteries. MSR-FUJI is equipped with two external electric sources for operation, and EDGs are required for backup. In case of station blackout (SBO: Loss of all AC electricity), it can be shut down and cooled without electricity.

9. Design and Licensing Status

Preliminary designs for various applications have been completed [3]. Three experimental MSRs were constructed, and one of them was operated for 4 years without severe problems. The detailed design has not yet started. Safety criteria and guidelines for MSR licensing are proposed [4].

10. Plant Economics

MSR-FUJI can achieve higher thermal efficiency owing to its high exit temperature. It has the potential to be operated for longer periods with a high availability factor since it does not require refueling shutdown. Also, fuel cycle advantages and cost saving can be realized due to its self-sustaining fuel cycle and no need for fuel element manufacturing. A plant construction cost for the 1000 MWe MSR-FUJI (super-FUJI) is estimated as less than $2000/KWe and the total electricity generation cost about 3 cents/KWh. In general the economic performance of a smaller plant may be expected to be worse than that of a larger plant. This factor has not been evaluated till now.

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[2] Yoshioka, R. “Nuclear Energy Based on Thorium Molten Salt”, Chapter-23 of the book “*Molten Salts Chemistry: From Lab to Applications*“ edited by F. Lantelme and H. Groult, Elsevier Inc., USA, 2013

[3] Yoshioka, R., Kinoshita, M. “Liquid Fuel, Thermal Neutron Spectrum Reactors”, Chapter-11 of the book “*Molten Salt Reactor*”, Elsevier Inc., USA, to be published in 2017

[4] Yoshioka, R., Mitachi, K., Shimazu, Y., Kinoshita, M. “Safety Criteria and Guidelines for MSR Accident Analysis”, *PHYSOR-2014*, Kyoto, Japan, 2014

Stable Salt Reactor (Moltex Energy, United Kingdom)

1. Introduction

The Stable Salt Reactor (SSR) is unique in its use of molten salt fuel replacing solid pellets in conventional fuel assemblies. This brings the major advantages of safe molten salts without the technical hurdles of managing a mobile liquid fuel. This combination should lead to substantially lower capital costs than conventional nuclear power technologies. The SSR is a design platform with two configurations, a thermal spectrum Low Enriched Uranium burner (SSR-U) and a fast spectrum actinide transmuting Wasteburning SSR (SSR-W). Both of these designs can be adapted in the future to breed fuel from thorium. In the thermal versions, the moderator is graphite and forms an integral part of the fuel assembly thus eliminating graphite lifetime concerns. The technology is coupled with large energy storage reserves to increase peak power demand allowing an increase of renewables on the grid. This summary focuses on the details of the thermal spectrum, uranium fuelled reactor SSR-U.

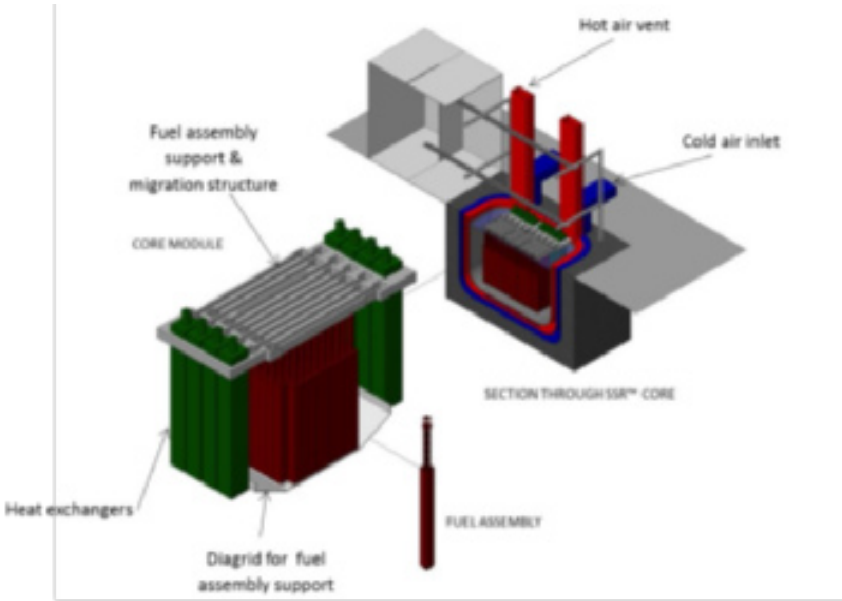


Figure 1: Isometric view of core module and section through reactor (Reproduced courtesy of Moltex)

2. Target Application

Stable Salt Reactor technology is initially aimed at mature markets with nuclear expertise. Once deployed and proven to be commercially viable the large market potential is in developing and non-nuclear nations due to the expected low cost and higher proliferation resistance. The high outlet temperatures further allows the plant to be coupled with thermal energy storage reserves in areas where there is grid instability due to fluctuating renewable electricity generation. This will allow a more stable energy market and an expansion of renewable power.

It is envisaged that one Wasteburning SSR-W will be required to burn the spent fuel and transmute the long lived actinides for ten thermal spectrum reactors. Extremely high levels of lanthanide contamination are permitted in the fuel of SSR-W. This enables a simple and economical pyroprocessing method to be used to convert virtually any spent fuel to fuel.

3. Development Milestones

2014	-UK patent granted for use of unpumped molten salt fuel in any reactor. -Independent capital cost estimate complete
2015	-Conceptual design complete and key claims validated by the UK's National Nuclear Laboratories.

4. General Design Description

Design Philosophy

The entire design philosophy is to reduce plant costs by simplifying the design and eliminating instead of containing hazards. This is done by combining the safety and operational benefits of molten salts along with conventional reactor components. Risks to the public are practically eliminated by design, and not merely contained.

The key features of the design are to achieve:

- Virtual elimination of the volatile radiotoxic source term under any conceivable accident, terrorist act or act of war. This would massively reduce the uninsurable liability (~$100 billion) currently born by national governments in the event of a serious release of radioactivity.
- Deployment of the SSR on smaller sites with smaller emergency planning zones as a result of this huge reduction in potential for offsite radioactive releases is expected to be approved by regulators. The probability that this is true is endorsed by the UK's National Nuclear Laboratory (NNL) [1].
- Economic competitiveness to have a capital cost similar to coal fired power stations but with substantially lower fuel cost. This is partly as a result of the intrinsic safety characteristics and partly due to the absence of pressurised reactor components – see details under Plant Economics.
- Modular design with the reactor assembled from road transportable, factory produced modules creating a single reactor unit of 300MWe. Larger plants of any size can readily be produced for larger demands which drop the Levelised Cost Of Electricity (LCOE) cost further.
- Fuel assembly form is compatible with IAEA Safeguards procedures used in reactors today (unlike other molten salt fuelled reactors).
- The output temperature of the reactor is high at 650-700°C. Thermal stresses are sufficiently low that the proposed standard stainless steels will not suffer creep degradation at these temperatures. The high output temperature has major advantages over the low output temperature of PWR reactors:
    - Use of standard low cost high efficiency superheated steam turbines now produced on a large scale for combined cycle gas turbine plants instead of larger, much less efficient saturated steam turbines which are now rarely produced except for nuclear reactors and which are commensurately more expensive
    - Heat storage using proven technology from solar thermal power plants is possible so electrical generation can be boosted during peak hours while the reactor runs at constant power. This allows carbon free load following to increase the capacity of renewables that the grid can accommodate.
    - SSR process heat has a wider range of commercial uses compared to low temperature nuclear steam.
- To design a reactor system that addresses both the long term sustainability of nuclear energy with the use of thorium and the problem of its resulting long lived radioactive waste.

Power Conversion Unit
A molten salt to steam boiler is proposed to generate steam. The turbine currently being specified for a 300MWe plant is the Siemens SST5-5000. The air cooled generator is Siemens SGen5-1200A-2P 118-55. Steam output temperature is 600˚C.

There will either be a larger turbine of 450MWe installed, to be coupled with the solar salt storage or two turbines of 300MWe and 150MWe. This will depend on the local electricity grid needs and economics.

Reactor Core
The core is made up of fuel assemblies which sit into a diagrid structure located at the bottom. At the top the assemblies sit in a rail system which forms part of the reactor 'lid'. The core is rectangular and the assemblies travel in rows laterally across the core. Each thermal spectrum assembly contains 37 vented tubes containing the fuel salt surrounded by a solid block of clad graphite as shown in *Figure 2*. The coolant passes around the tubes through the centre of the graphite.

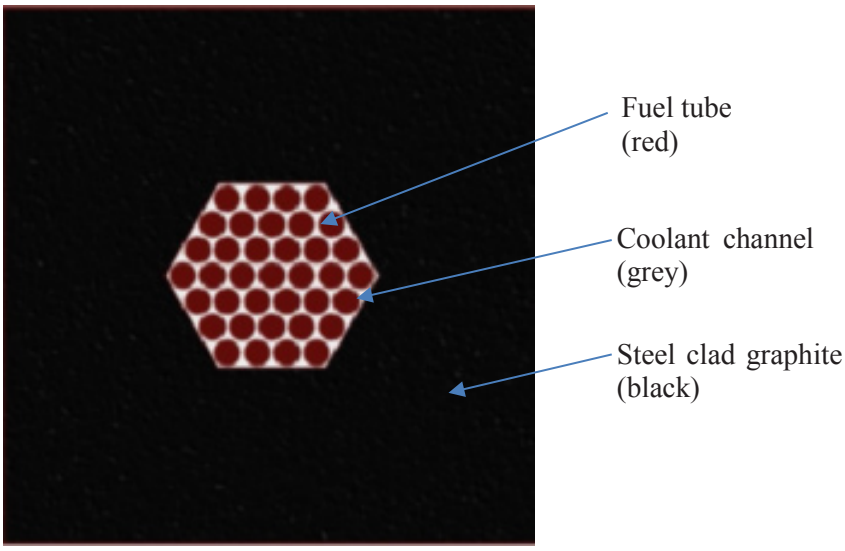
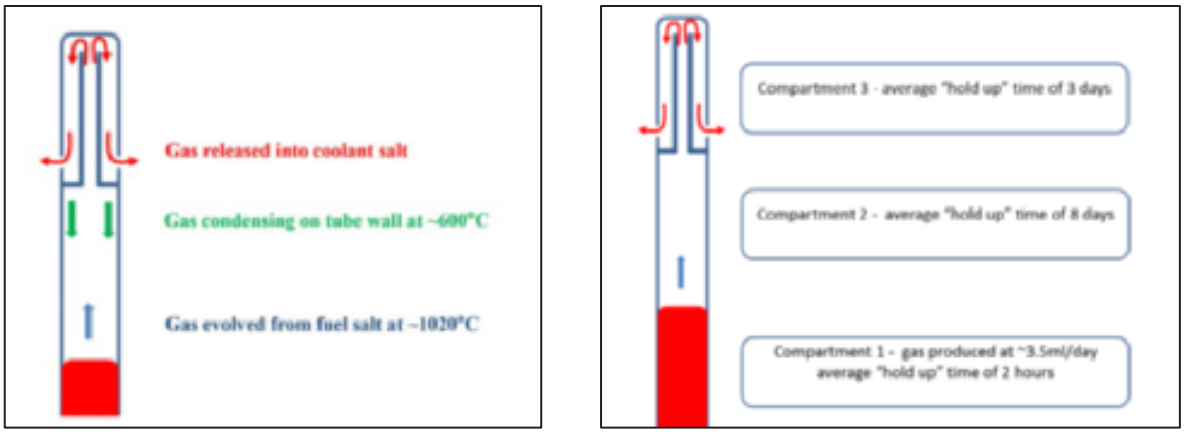


Figure 2: Section through fuel assembly showing the graphite, fuel tubes and coolant flow channel (Reproduced courtesy of Moltex)

The tubes are stainless steel, similar in materials and structures to today's AGR fuel. The tubes are 3m long with 2.6m of fuel and 400mm of gas space at the top which has a venting mechanism as shown in *Figure 3* to allow some gaseous fission products to be released into the coolant salt (most radiologically important fission products are captured in a non-gaseous form). In addition to preventing build-up of pressures within the tubes, the venting mechanism walls are cooler than the gas itself which causes it to condense on the walls and flow back into the fuel. Fission product gases have time to decay to more stable isotopes before entering the pool of coolant salt. Most released fission product gases are furthermore absorbed by the coolant into stable salt forms. Only the noble gases and cadmium migrate through the coolant salt to the argon containment zone above.

Fuel Characteristics
The thermal spectrum fuel is a molten fluoride salt with a eutectic composition of 33%NaF / 30%RbF / 37%UF4 and a melting point of 535˚C. The fuel is prepared in a simple single step reaction from uranium hexafluoride, as is already produced in many parts of the world. Fuel pellets are melt cast from this mixture requiring none of the complex sizing, sintering and

machining required for oxide fuel. SSR fuel therefore has the potential to be substantially cheaper than today's fuel. Once the assembly is lowered into the tank of molten coolant salt, the fuel pellets within the tubes will melt. Only then, will they be taken into the core where they will become critical.



Figures 3: Venting mechanism at the top of each fuel tube causes gaseous fission products to condense on the walls ensuring a hold up time before it is released to the coolant salt beyond the tube (Reproduced courtesy of Moltex)

Fuel Handling System

The rectangular core enables lateral fuel shuffling through a rail system without lifting the assembly vertically out of the coolant during a fuel change. The vessel does not need to be flooded as it does in a PWR. The assemblies in alternate rows are moved in opposite directions during reload. This result in a pattern where if fresh fuel is inserted in one end, the two assemblies either side of it are depleted assemblies ready to be removed - they were initially loaded as fresh assemblies from the other side of the core. A new fuel assembly is added to a row every 6 months. At the same time, the assembly at the other end is removed. New assemblies are added to different rows every few days. This is similar in strategy to CANDU fuel shuffling and results in a sufficiently flat power profile across the core.

Once an assembly leaves the core it sits at the edge of the tank until it has cooled sufficiently to be lifted into the argon space above the core to cool down. The fuel quickly solidifies due to the high temperature gradient between itself and the argon. It can then be removed from the reactor and placed into dry cask storage until ready for reprocessing or geological disposal.

Reactivity Control

No excess reactivity needs to be added to the core to compensate for fuel burn up because the combination of frequent on power refuelling and high negative temperature reactivity coefficient allow the core to generate constant power between refuelling steps – the small drop in reactivity is compensated by a small fall in average fuel salt temperature, which is readily compensated by changes in the coolant flow rate.

No reactivity shims or control rods are required at any time under normal operating conditions, eliminating the potential for control failures that can lead to an increase in the core reactivity. The excess reactivity always required at start up is provided either by step by step addition of the final fuel assemblies with the coolant at normal operating temperature or by adding all the fuel assemblies with the coolant at a higher temperature and then carefully lowering the temperature. A combination of the two methods is also practical.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	Moltex Energy
Country of origin	United Kingdom
Reactor type	Static Fuelled Molten Salt Reactor
Electrical capacity (MW(e))	300MWe (formed by eight 37.5MWe modules)
Thermal capacity (MW(th))	750MWth
Expected capacity factor (%)	91%
Design life (years)	60 years
Plant footprint (m^2)	22500
Coolant/moderator	Graphite as an integral part of fuel assembly
Primary circulation	Forced circulation
System pressure (MPa)	Atmospheric
Core inlet/exit temperatures (^oC)	550˚C - 700˚C
Main reactivity control mechanism	Negative temperature coefficient (net of fuel plus coolant is -5.4pcm/K)
Tank height (m)	4m (H)
Tank Size (m)	18m (L) x 5m (W)
Module weight (metric ton)	<10t per 37.5MWe module excluding salt
Configuration of reactor coolant system	Pool type
Power conversion process	Steam generator to turbine
Cogeneration / Process Heat Capabilities	Yes, possible.
Passive Safety Features:	Large negative temperature coefficients, decay heat removal by passive air cooling ducts, volatile source term reduced by 10^6 in accident scenario (due to the lower partial vapour pressures of volatile fission products in a severe accident with molten salt fuel compared to a uranium oxide fuel pellet).
Active Safety Features:	Shutdown blade SCRAM + fluoroborate poison
Fuel type/assembly array	Molten salt fuel within vented fuel tubes in a conventional style fuel assembly within graphite.
Fuel assembly active length (m)	2.6m
Number of fuel assemblies	1,040 (per 300MWe)
Fuel enrichment (%)	<15% LEU
Fuel burnup (GWd/ton)	>150GWd/tHM
Fuel cycle (months)	~84 months
Approach to engineered safety systems	Reactor has multiple inherent safety features. Inherent features will be given a low probability of failure in the Probabilistic Safety Assessment.
Number of safety trains	Nominally 4 way segregation where applicable
Emergency Safety Systems	Passive Air Cooling Ducts around the tank walls; multiple diverse SCRAM
Residual Heat Removal System	Thermal inertia of tank by natural circulation and passive air cooling ducts.
Refuelling outage (days)	On line regular refuelling – maintenance outage under review.
Distinguishing features	Molten salt fuel constrained in conventional, vented fuel tubes. Thermal energy storage to increase peak electrical capacity.
Modules per plant	8 per 300MWe (no upper limit to number of tanks)
Target construction duration (months)	31 months from start of nuclear concrete
Seismic design	Yes, suspended tank on seismic fixings.
Core damage frequency (per reactor-year)	Expected $<10^{-6}$ (and the consequence / hazard to the public reduced by 10^6)
Design Status	Concept, Preparing licensing pack

Reactor shutdown is provided by shutdown blades with electromagnetic securing of the control blades. The backup method is a coolant neutron poison system based on sodium fluoroborate which can be released based on a thermally triggered system.

Reactor Containment System
There are no pressurised systems or components within the reactor building. The tank is a stainless steel vessel suspended using anti-seismic suspension fixings. The tank is cooled by air cooling ducts around the perimeter. A thin walled stainless steel liner surrounds the wall and roof of the argon containment zone. This is surrounded by a ~1m blast resistant concrete wall which also acts as a biological shield.

A large airlock allows core modules, fresh and spent fuel assemblies to be moved in and out of the core. Argon venting is done irregularly through the airlock.

Reactor Coolant System
The reactor is a pool type reactor so the primary coolant fills the tank. Use of an RbF based fuel salt means that the coolant must operate between 570°C and 700°C in order to prevent fuel salt freezing. The coolant salt is therefore 39% ZrF_4 / 1% ZrF_2 / 60% NaF with a melting point 500°C.

A secondary coolant salt with additional potassium fluoride to lower the melting point is used to take the heat from the primary heat exchanger to the boiler tubes. Its composition is 10% NaF / 48% KF / 42% ZrF4 and it has a melting point of 385°C. The solar salt thermal tanks are heated through a direct loop into the solar salt tanks. Initial heating of the water/steam from the condenser is done with a small heat exchanger of solar salt to bring the steam temperature to above 435°C.

Steam Generator
The boiler tube design is under review but is being designed using knowledge gained from the concentrated solar industry. Due to the high melting point of the secondary coolant (385°C), a solar salt of lower melting point is used to do the initial heating of the water/steam from the condenser before the steam can enter the main steam generator. This is to ensure there is no freezing of the secondary coolant fluoride salt on the boiler tube walls.

5. Safety Features

Engineered Safety System Approach and Configuration
The molten salts used in the SSR are chemically stable with minimal reactions with air or water. Specific design solutions and material choices prevents corrosion of the fuel clad (with fuel salt) and plant components with the coolant salt. Use of molten salt fuel with the correct chemistry further eliminates the hazardous volatile iodine and caesium source terms which prevents airborne radioactive plumes in severe accident scenarios.

Pool type reactor designs are well established as particularly safe options since natural convection of heat from the core is sufficient to remove decay heat. This design element has been incorporated into the SSR. However, the coolant is chemically non-reactive, under no pressure and has an extremely high boiling point making the pool design even safer than those used in other reactors. This single factor separates the SSR from all other molten salt reactor designs where relatively complex engineered systems are required to assure continued cooling of the fuel.

The non-volatile radioactive fission products are physically contained within the fuel tubes.

In the unlikely event of tube rupture the coolant salt has been selected so as to be miscible with the fuel salt. If multiple tubes are ruptured, the large tank of coolant provides a means of massively diluting the fuel rendering the reactor subcritical while maintaining its containment within the reactor tank without relying on drain tank valves.

Below ground construction - The reactor tank is located below ground level in a steel and concrete lined pit. This avoids any potential for a physical loss of coolant sufficient to expose the core and provides protection (in conjunction with above ground concrete shielding) against aircraft impact or other external factors.

Decay heat removal system
Natural convection of the primary coolant salt will continue in the event of a reactor shutdown or pump failure. The primary coolant heats up to a point where radiative heat transfer from the tank walls becomes the dominating factor. A finned air duct to atmosphere exists around the tank walls which can take decay heat away in this accident scenario.

Emergency core cooling system
The core, heat exchangers and pumps are designed so that natural circulation can continue indefinitely. This ensures the core is continually cooled.

Containment system
See '*Reactor Containment System'* under Section 4 above. There are no internal pressures. The primary containment is the tube wall, secondary is the coolant salt (which absorbs fission products). The third is the tank itself and the fourth is the concrete structure. Above the tank is an argon space which has a stainless steel liner surrounded by a ~1m concrete wall which serves as the biological shield. The concrete above and below grade is designed to be blast resistant. The argon containment zone is at a lower pressure than the air within the reactor building to minimise gas escapes in the event of a breach. However, the composition of the gas within the containment is maintained such that a major release due to massive containment loss would not release hazardous quantities of radioisotopes. The reactor building walls are ~300mm thick reinforced concrete. This serves as building structure and as an initial energy reducer from an external impact of a missile. The building will be aircraft resistant although the consequence of an aircraft impact is substantially lower than in a PWR due to the reduced volatile source term.

6. Plant safety and Operational Performances

The design philosophy is such that no operator access is ever required in the main reactor zone. The substantial reduction in quantity of engineered safety and component systems will substantially reduce the number of operating staff required. The ramp rates of the plant will be driven by the steam side, not the nuclear side. Shutdown and maintenance periods will also be driven by the requirements of the steam turbine.

7. Instrumentation and Control systems

Primary reactivity control will be by the reactivity coefficient of the coolant and fuel. Heat removal driven by the turbine will control the reactivity sufficiently. There will be neutron and temperature sensors above the core area and within the coolant. All components are designed with the facility to be inspected remotely by visual or mechanical means.

There will be radiation and chemical monitoring of the gas space to ensure no unexpected fission products are being released to atmosphere (in the event of a leak or opening of the air lock).

8. Plant Arrangement

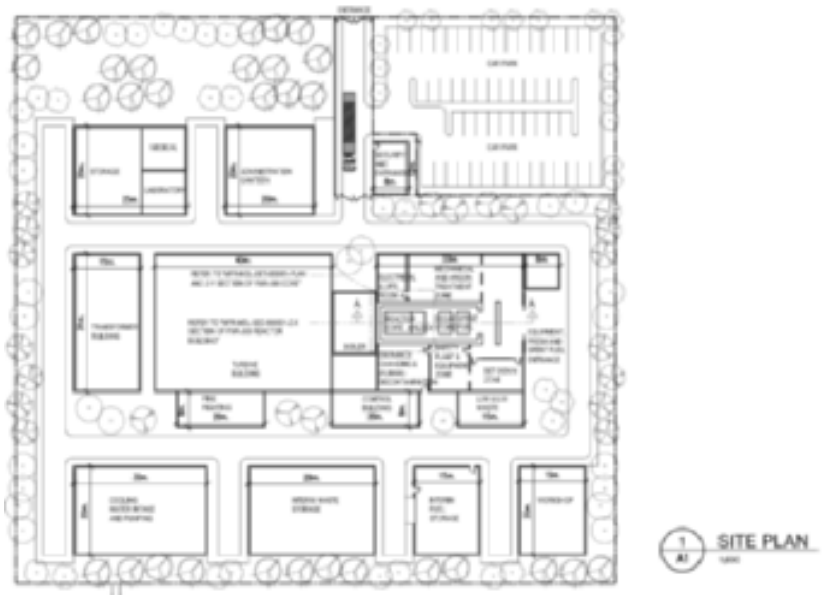


Figure 4: Illustrative site layout for a 300MWe plant (Reproduced courtesy of Moltex)

9. Design and Licensing Status

The focus to date has been on the novel aspects of the reactor including neutronics and chemistry. The majority of the design work has been carried out by various UK consultancies and universities. The current phase of design is at component level where supplier engagement is taking place to specify each component. The Preliminary Safety Case is in preparation to initiate licensing in 2017.

10. Plant Economics

Partly as a result of the intrinsic safety and partly due to absence of pressurised reactor components the overnight cost of the complete nuclear island, including civil works, of a 1GW fast version of the SSR has been estimated independently of Moltex Energy by Atkins Ltd as being £714 per kW ($1,071/kW) capacity. This costing has been used as the basis for estimating full plant cost for a 1GWe thermal spectrum SSR-U of £1,892 per kWe ($2,838/kW). These are based on UK on-site construction costs, potential cost savings for modularised construction is not included. Substantial prelims and contingencies are included. This provides a highly credible case for reactors having comparable capital cost to coal fired power stations but with radically lower fuel cost. Levelised electricity costs are not given at this stage as they are so heavily dependent on the specific financing mechanisms.

The far simpler design of the Stable Salt Reactor compared to PWRs and AGRs and small number of safety critical systems should lead to significantly lower O&M costs. The process for converting enriched uranium hexafluoride into fuel ready for incorporation into fuel assemblies is very much simpler for Stable Salt Reactor fuel than for oxide fuel and is also therefore expected to be significantly cheaper. Subsidies will not be required as the electricity produced by the SSR will be competitive with fossil fuels even without carbon tax.

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[1] Review of Key Claims of the Stable Salt Reactor, NNL plus other Subject Matter Experts (report available under confidentiality), 2016.

Further information can be found on line at www.moltexenergy.com

SmAHTR (Oak Ridge National Laboratory, United States of America)

1. Introduction

SmAHTR is a deliberately small (125 MWt) fluoride salt–cooled high-temperature reactor (FHR) design concept intended to match the energy requirements of coupled industrial processes. SmAHTR's overall purpose is to enable an integrated assessment of the potential performance of deliberately small FHR designs and to provide guidance to the overall FHR research and development effort. SmAHTR's development effort has been in support of the US Department of Energy (DOE) Office of Nuclear Energy's Advanced Reactor Technologies Program. SmAHTR is an early phase concept and therefore is not commercially viable at this time.

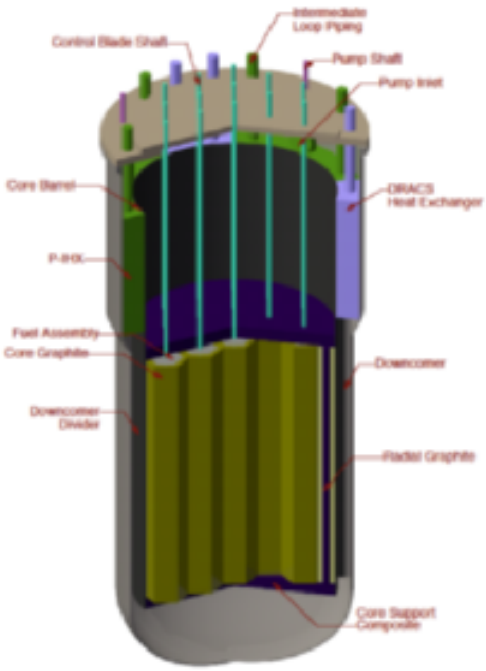

Figure 1: SmAHTR in and near vessel structures (Reproduced courtesy of ORNL)

FHRs are an emerging reactor class that combines attractive attributes from previously developed reactor classes and power plants. FHRs by definition feature low-pressure liquid fluoride salt cooling; high-temperature-tolerant, salt-compatible fuel; a high-temperature power cycle; and fully passive decay heat rejection. FHRs have the potential to economically and reliably produce large quantities of electricity and high-temperature process heat while maintaining full passive safety. Leveraging the inherent reactor class characteristics avoids the need for expensive, redundant safety structures and systems and is central to making the economic case for FHRs. Additionally, as a high-temperature reactor class, FHRs can efficiently generate electricity and provide the energy for high-temperature industrial processes (notably including the production of hydrocarbon fuel). Moreover, high-temperature operation increases FHR compatibility with dry cooling. *Figure 1* shows the layout of the SmAHTR in-vessel structures.

2. Target Application

The particular variant of SmAHTR described in this report is designed for efficient hydrogen

production using the carbonate thermochemical cycle (CTC) [1] The SmAHTR-CTC design variant is an evolution of the SmAHTR design concept first described by Greene et al. in 2010 [2]. Additional detail about the design is available in ORNL/TM-2014/88.

3. Development Milestones

2010	Initial SmAHTR conceptual design developed
2014	Perform refined SmAHTR concept study focused on high-temperature heat production

4. General Design Description

Design Philosophy

The "new build" strategy being pursued for SmAHTR-CTC has three guiding principles: (1) leverage plant characteristics for maximum economic performance, (2) maximize reliability, and (3) minimize licensing risk. The design intent is to leverage the inherent reactor characteristics to drive down cost while to the extent possible staying within the structure of the current US Nuclear Regulatory Commission (NRC) licensing framework. A key element of the strategy is to rely upon SmAHTR's strong, inherent safety characteristics to avoid the need for expensive, redundant safety structures and systems.

Power Conversion Unit

SmAHTR's hydrogen production plant will implement the CTC. The peak cycle temperature is not greater than 650°C, and the peak pressure is at most a few atmospheres. In the CTC process, uranium valence changes drive the decomposition of water to hydrogen and oxygen. The key innovative step is reacting triuranium octoxide (U_3O_8) with sodium carbonate (Na_2CO_3) and steam to generate hydrogen and sodium diuranate ($Na_2U_2O_7$) at ≤650°C [3]. The reaction is suitable for implementation in a screw calciner. The relatively narrow and variable energy gap, between uranium's 6*d* and 5*f* atomic orbitals, enables uranium to assume more than one valence state under relatively mild temperatures and pressures. Shifting uranium's oxidation state liberates the hydrogen from the steam. The cycle process steps necessary to regenerate U_3O_8 and Na_2CO_3 from $Na_2U_2O_7$ are already industrially implemented in the uranium processing industry. However, the CTC to date has only been demonstrated at laboratory scale.

Reactor Core & Fuel Characteristics

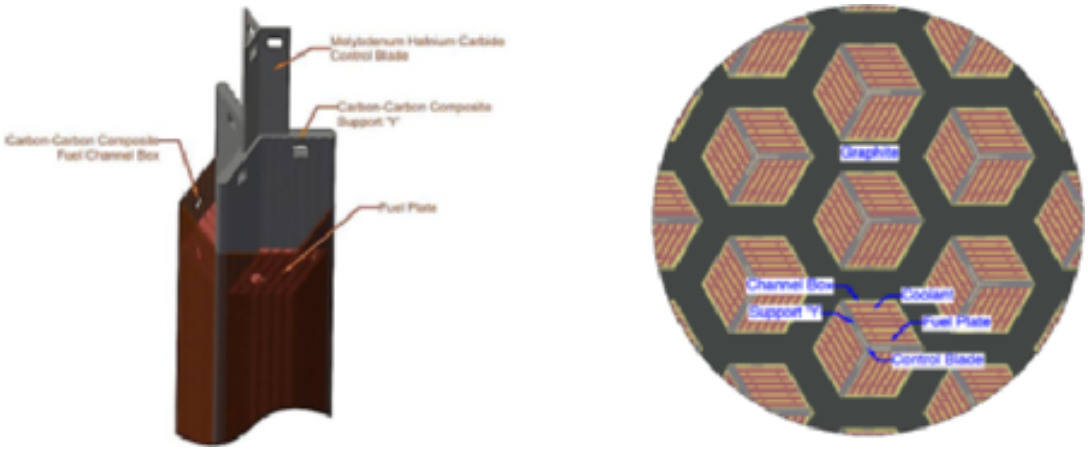

Figure 2: SmAHTR fuel assembly upper end and partial core cross section (Reproduced courtesy of ORNL)

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	ORNL
Country of origin	USA
Reactor type	Molten Salt Reactor—FHR
Electrical capacity (MW[e])	NA
Thermal capacity (MW[th])	125
Expected capacity factor (%)	>90%
Design life (years)	60 years
Coolant/moderator	FLiBe / carbon
Primary circulation	Forced circulation
System pressure (MPa)	Atmospheric
Core inlet/exit temperatures (°C)	670/700°C
Main reactivity control mechanism	Negative temperature coefficient; control blade insertion
RPV height (m)	9
RPV diameter (m)	3.5
RPV or module weight (metric ton)	22.5 (empty, no lid)
Configuration of reactor coolant system	Integral
Power conversion process	Carbonate thermochemical cycle
Passive safety features:	Large thermal margin, multiple fission product barriers, multiple barriers to uncovering fuel, strong negative temperature reactivity feedback, thermally triggered shutdown mechanisms, slowly evolving accidents, passive natural circulation-based decay heat rejection, low-consequence fuel handling accidents, structurally robust below-grade nuclear facilities, non-water soluble fuel, reactor isolation from non-nuclear facilities
Active safety features:	Not necessary
Fuel type/assembly array	Plate TRISO
Fuel assembly active length (m)	4
Number of fuel assemblies	19
Fuel enrichment (%)	8
Fuel burnup (GWd/ton)	69
Fuel cycle (months)	6
Residual heat removal system	Direct reactor auxiliary cooling system (DRACS) 2 out of 3
Refuelling outage (hours)	8
Design status	Pre-conceptual design

SmAHTR uses tri-structural isotropic (TRISO) coated-particle fuel embedded near the surfaces of carbon plates. SmAHTR uses the uranium oxycarbide TRISO fuel particles currently being tested under DOE's advanced gas reactor fuel development program. The fuel particles are located near the surfaces of the plates to lower the peak particle temperature by improving the thermal coupling to the coolant. The plates are configured into hexagonal assemblies with 18 plates per assembly. SmAHTR's fuel assemblies will closely resemble shortened versions of those for the larger advanced high-temperature reactor (AHTR) [4]. The fuel assemblies will be mounted together into a cartridge core using continuous-fiber-composite core support plates. SmAHTR's cartridge core enables filling of the inter-assembly volume with nuclear-grade graphite, resulting in improved neutron utilization, as shown in *Figure 2*. The entire core will be lifted out as a single unit for refueling. Each fuel assembly

includes a molybdenum hafnium carbide control blade. While plate-style TRISO fuel bodies have not been manufactured previously, the manufacturing process steps of any geometric-shape fuel body are nearly identical.

Fuel Handling System

SmAHTR-CTC's transparent, low-pressure coolant enables refueling to be performed rapidly using conventional mechanical manipulation technology that is visually guided from above. To refuel, the reactor is first brought subcritical by inserting the control blades. Once the blades have been inserted, the reactor vessel lid can be removed. The core can then be moved via an overhead crane system into the used fuel pool. The thermally robust nature of coated-particle fuel significantly eases the fuel transfer safety design requirements. Even just a few hours after shutdown, the core can withstand more than 15 minutes of removal from liquid cooling, during the fuel transfer process, without exceeding its allowable maximum temperature. SmAHTR-CTC uses a shielded fuel transfer ramp mechanism conceptually derived from the Phénix SFR fuel-handling system. Once the reactor vessel top has been removed, a shielded ramp is positioned (using a rail system) with its entrance over the top of the reactor vessel. The core is raised out of the vessel and up the ramp using an overhead crane; and upon reaching the top of the ramp, it is allowed to swing to its opposite side. The core is then lowered into the used fuel pool, sliding down the far side of the ramp. The core-lifting mechanism includes a thermal fuse, so that if at any point during the fuel transfer the maximum allowable temperature were exceeded, the core would be passively released to slide downward into a cooled configuration. SmAHTR-CTC employs a twin cartridge-core configuration. A cartridge configuration enables the core to be moved as a single unit, minimizing the number of mechanical manipulations required for refueling. In a twin core system, one core is in the reactor vessel and the other is in the used fuel pool. The twin core configuration enables assembly-level fuel shuffling to be performed while the alternate core is in operation. SmAHTR's core needs to be moved as a single unit because of the tight clearances of its structural elements. Some radiation-induced mechanical distortion will occur during operation. The mechanical distortion will prevent the removal of individual fuel assemblies from the top. The core internal graphite pieces will be replaced as the fast neutron fluence causes excessive mechanical distortion (every few years). Separating the fuel from the graphite moderator enables separate disposal of the in-core graphite and the fuel assemblies, decreasing the volume of high-level waste. SmAHTR-CTC maximizes its neutronic efficiency by maximizing the amount of carbon in the core while minimizing the amount of FLiBe.

Reactivity Control

SmAHTR has two primary shutdown mechanisms: (1) control blades and (2) poison salt injectors. SmAHTR does not have any threshold phenomena that would occur if the primary coolant temperature were to exceed the design point by several tens of degrees for hundreds of hours. Consequently, the temperature rise can be employed to passively initiate shutdown. If SmAHTR's core outlet temperature were to exceed its design limits, the control blades would be passively inserted as a result of melt point fuses in the control blade linkages. The control blades are jointed to provide sufficient flexibility to enable insertion if their guide tubes become distorted. The control blades are also actively engaged to the drive motors. Upon loss of power, the control blades would disengage from the drive mechanisms and drop into the core. The poison salt injectors have a single-ended cylinder and plunger format. Pressurized inert gas is maintained on the closed end of the cylinder. A sliding disk divides the cylinder volume into open and closed ends. The open end of the cylinder is filled with poison salt powder (EuF_3 or GdF_3). A melt point alloy holds a lid onto the open end of the

cylinder during normal operations. The poison salt accumulator is located in SmAHTR's lower plenum. If the lower plenum temperature exceeds the set point value, the melt point trigger releases and the poison salt is injected into the core inlet. The melt point release mechanism can also be triggered by the plant operators via embedded electrical heaters. Both the primary and secondary shutdown mechanisms provide substantially more negative reactivity worth than would be necessary to shut down the reactor. The reactivity worth of the control blades is sufficient to shut down the reactor with the most reactive blade stuck out of the core at all points during the fuel cycle. As the intended refueling interval is only 6 months, blade insertion capability will be confirmed frequently. The solubility of the reserve shutdown poison salts is much greater than the potential concentration in the lower vessel plenum, providing confidence that the salts would dissolve completely in the primary coolant and rapidly bring the reactor subcritical.

Reactor Pressure Vessel and Internals

SmAHTR employs an Alloy 800H reactor vessel with an Alloy N lining. The fueled core and the radial reflector are contained within a 2-cm-thick core barrel made of carbon-carbon composite that separates the up-flow from the downcomer. The inside of the core barrel facing the core has a thin plating (1 cm) of boron carbide to reduce the neutron flux on the reactor vessel. The downcomer is sized so that its transverse flow area is at least twice the in-core flow area.

5. Safety Features

The safety characteristics of FHRs arise from fundamental physics and well-designed, constructed, and maintained structures, systems, and components (SSCs). Until an approved set of reactor-class-specific general design criteria, as well as design basis accident (DBA) sequences, is available, a combination of engineering judgment and the safety characteristics of FHRs will be relied upon to guide SmAHTR-CTC design. FHRs feature full passive safety and do not require any active system or operator response to avoid large off-site release for any DBA or evaluated beyond-DBA, including earthquake, tsunami, large commercial plane impact, or permanent station blackout. More specific information about SmAHTR-CTC's safety characteristics is provided in the following paragraphs.

Large Thermal Margin To Fuel Failure

The silicon carbide barrier layer of properly operating coated-particle fuel will remain intact several hundred degrees above SmAHTR-CTC's peak fuel temperature, retaining nearly all of the fission fragments. The primary FLiBe coolant will not boil until it reaches temperatures over 1400°C. SmAHTR's core outlet temperature is 700°C. Additionally, the viscosities of fluoride salt coolants decrease with increasing temperatures, resulting in increased coolant flow to the hottest fuel, thereby reducing fuel hot spots. Further, while the power density of FHR cores is above that of high-temperature gas-cooled reactors (HTGRs), FHRs contain larger numbers of fuel particles, resulting in low average particle power (26.5 mW/particle for the SmAHTR-CTC compared with ~150 mW/particle for the next-generation nuclear plant HTGR design). This limits the radiochemical challenge to the silicon carbide containment layer within the fuel.

Multiple Fission Product Barriers

As described in the previous section, SmAHTR-CTC features four independent containment layers along with a substantial radionuclide source term reduction due to the core's immersion in fluoride salt. Additionally, the low system pressure prevents the development of a significant driving force, which would cause the dispersion of radionuclides in the event of barrier failure.

Multiple Barriers To Prevent Uncovering Fuel
The reactor vessel does not include any penetrations below ~20 cm above the top of the direct reactor auxiliary cooling system (DRACS) heat exchangers. Fluoride salts have high volumetric heat capacity, and a large volume of salt is maintained over the core, providing a substantial thermal margin. Further, owing to the high melting point of the primary salt coolant, small leaks in the reactor vessel are expected to self-seal as the coolant freezes upon leaking. Although the fuel is slightly buoyant in the coolant, multiple upper core support structures, including the upper core support plate, the weight of the control blade, and the control blade guide tube, all maintain the fuel below the surface.

Strong Negative Temperature Reactivity Feedback
The strong overall reactor negative reactivity temperature feedback will prevent reactivity insertion accidents from causing fuel damage. The negative Doppler fuel temperature coefficient for fresh fuel is similar in magnitude to that of a typical pressurized water reactor. The coolant temperature feedback coefficient is also negative, but it is one order of magnitude smaller than the negative Doppler feedback, while the moderator temperature coefficient is near zero. The >700 K margin to coolant boiling, combined with the deep coolant pool, makes significant voiding of the core highly improbable. The combination of void and temperature coefficients is always negative under these circumstances, thus avoiding the potential for fuel damage reactivity accidents. Bubbles within the primary coolant flow (caused by, for example, gross failure of the reserve shutdown system piping) would be a DBA that would cause local fuel heating as the bubbles rise through the core. However, the large thermal margin to fuel failure, the rapid Doppler feedback, the limited duration of bubble passage, and the longer neutron path length in graphite-moderated cores alleviate the potential for bubble entrainment to damage the fuel. Additional safety evaluations are necessary to better assess the heat transfer consequences of the events under this scenario.

Thermally Triggered Shutdown Mechanisms
SmAHTR-CTC's most thermally sensitive component is its reactor vessel. Exceeding the reactor vessel's design temperature for long periods of time would increase its creep. However, creep is not a threshold process, and SmAHTR-CTC's reactor vessel creep budget would allow for hundreds of hours of operational time at several tens of degrees above normal. The ability of the vessel, core, and coolant to tolerate temperature rises enables the use of thermal triggers to provide negative reactivity feedback. SmAHTR-CTC's control blades feature thermal fuses (melt point alloys) located in the upper plenum that are designed to release the blades in the event of an over-temperature accident without SCRAM. Also, SmAHTR's secondary shutdown poison salt accumulators will be set to inject poison salt into the core inlet plenum in the event of more-severe temperature excursions.

Slowly Evolving Accidents
SmAHTR-CTC has no credible primary coolant-phase-change accidents. SmAHTR-CTC also has a much lower power density core than modern light water reactors (LWRs) (~5.3 MW/m^3 vs ~110 MW/m^3), resulting in significantly slower transient response. The power density is calculated to be 9.4 MW/m^3 if the carbon between the fuel assemblies is not considered to contribute to the core's volume (referred to as "power density in fuelled volume"). Further, the large volume of salt (the volumetric heat capacity of FLiBe is 4670 kJ/m^3-K at 700°C) and graphite in the reactor vessel provides for substantial local thermal storage to provide time for recovery operations.

Passive, Natural Circulation-Based Decay Heat Rejection

SmAHTR employs natural circulation DRACS-type decay heat rejection to the local air. Fluoride salts have large coefficients of thermal expansion that, combined with their large volumetric heat capacities, enable excellent natural circulation cooling. Similar natural-circulation-driven emergency heat rejection is also provided to the used fuel storage pool. The DRACS are modular and independent, enabling the system to continue to function following the failure of individual units. SmAHTR-CTC's design calls for three independent DRACS and is intended to withstand DBAs with only two out of three systems in operation. Sufficient decay heat rejection is provided to avoid vessel failure in the event of complete, permanent station blackout accompanied by the loss of a single DRACS. In addition, SmAHTR-CTC will not require a dual interconnection to the grid, as the plant does not depend on the availability of electrical power to reject decay heat. SmAHTR-CTC does not require an emergency core cooling system or high-volume, safety-grade coolant makeup systems. Significantly lowering the primary coolant level in the vessel requires both primary and guard vessel failures, as no other mechanisms have been identified to remove substantial amounts of coolant from the vessel.

Low-Consequence Fuel-Handling Accidents

SmAHTR-CTC's core is slightly buoyant. However, each fuel assembly includes a molybdenum alloy control blade to provide sufficient ballast to sink the fuel in the event that the core is dropped during movement. Liquid fluoride salts are transparent, enabling fuel movement to be visually guided, thus substantially reducing the risk of unanticipated mechanical impacts while enabling visual confirmation of mechanical motion. Furthermore, since only the weighted core sinks slowly in the coolant, mechanical damage to the core or any other reactor structures as a result of its being dropped would not be anticipated.

Structurally Robust, Below-Grade Nuclear Facilities

FHR buildings and used fuel storage pools are intended to be located below grade, providing shielding from aircraft impact and wind/waterborne missiles. The DRACS cooling towers are spatially separated to avoid simultaneous failure of both as a result of a single aircraft impact and are structurally robust to minimize their vulnerability to likely natural events (e.g., large tornados or earthquakes).

Non-Water-Soluble Used Fuel

Carbon-enveloped coated-particle fuel is not water soluble, enabling used fuel assembly emplacement in shallow (<1 km) dry wells for eventual (after 6–12 months in an in-containment used fuel pool), local, intermediate-term, retrievable storage with a very low probability for radionuclide escape into the surrounding environment. The first layer of containment remains with the fuel after it is unloaded from the core, and the silicon carbide containment layer is not anticipated to fail in the more benign used fuel storage environment. The envisioned local used fuel storage dry wells (boreholes) closely resemble cased and cemented petroleum wells, which can effectively reject decay heat to the surrounding rock as a result of their large surface-to-volume ratio.

Reactor Isolation From Non-Nuclear Facilities

SmAHTR-CTC's low-pressure liquid-salt intermediate loop prevents accidents in the CTC from adversely impacting the nuclear plant. The relatively low cost of the intermediate loop per unit of length enables separation of the reactor building from the hydrogen production facility by substantial distances (>100 m). The reactor will also be separated from potential pressure waves propagating down the intermediate loop piping by rupture disks, or other conventional pressure relief mechanisms, so that it is not credible for a disturbance (including hydrogen explosions) within the CTC to propagate to the nuclear island.

6. Decay Heat Removal

SmAHTR-CTC's normal core decay heat removal is via the intermediate cooling loop to the balance-of-plant heat rejection system. During maintenance shutdowns, SmAHTR-CTC's core will be removed to the used fuel cooling pool. Decay heat will be removed via the three independent DRACS cooling loops under loss-of-forced-flow accident conditions. The DRACS cooling loops transfer heat from the DRACS heat exchangers located within the reactor vessel to the natural draft heat exchangers (NDHXs) located at the bases of chimneys external to the reactor building. Each NDHX is located within its own chimney to avoid common vulnerability to a single external impact. During normal operation, each NDHX is contained within an insulated shell to minimize the parasitic heat loss, and resultant potential for freeze-up, and to enable tritium capture within the NDHX shell. Each DRACS loop is sized to provide ~0.5% of full-power heat rejection under fully developed flow conditions at 700°C. Only two of the three DRACS loops are required to protect against an unacceptable temperature rise following or during a protected loss-of-forced-flow accident. The ultimate heat sink is outside air. Used fuel pool normal-condition cooling is provided via a pumped salt loop to a forced convection heat exchanger coupled to the outside air. Safety-grade heat rejection is provided for the used fuel pool through a two-phase, reduced-pressure, liquid-metal thermosyphon tube wall with the hot ends of the tubes dipped into the used fuel pool and the condenser section above the top of the containment structures.

7. Plant Economics

SmAHTR-CTC's objective is to economically and safely produce hydrogen and thereby maximize the return on investment to the plant owners. A key element in ensuring advantageous economic performance is minimizing the risk premium element of the interest rate paid during construction. The interest rate risk premium is determined based upon how confident a lender is that an FHR plant will (1) be constructed at planned cost and schedule, (2) obtain regulatory approval to operate, and (3) operate reliably for many years. SmAHTR-CTC's design is not yet mature enough to develop a "bottom-up" cost model. Consequently, the economic evaluation is limited to comparison with similar technologies. Nearly all aspects of FHRs will either be less costly than or have costs equivalent to those of LWRs. The major exception to the comparative cost advantage is the markedly higher cost of the primary coolant. Initial FHRs will also incur significant first-of-a-kind expenses. For example, SmAHTR's initial fuel load of TRISO coated-particle fuel will cost much more than an equivalent LWR fuel because of the need to construct new manufacturing facilities. SmAHTR is a high-temperature reactor, which would enable it either to have higher thermal efficiency for electricity production or to produce high-temperature process heat. SmAHTR-CTC's design is optimized to provide high-temperature process heat, avoiding initial competition with mature technology.

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[1] J. L. Collins, et.al., *Carbonate Thermochemical Cycle for the Production of Hydrogen*, US Patent 7,666,387, B2, February 23, 2010.

[2] S. R. Greene, et.al., *Pre-Conceptual Design of a Small Modular Fluoride Salt-Cooled High Temperature Reactor (SmAHTR)*, ORNL/TM-2010/199, December 2010.

[3] J. Ferrada, et.al., "Carbonate Thermochemical Cycle for the Production of Hydrogen," National Hydrogen Association Annual Conference and Hydrogen Expo, Columbia, South Carolina, March 30–April 3, 2009.

[4] V. K. Varma, et.al., *AHTR Mechanical, Structural, and Neutronic Preconceptual Design,* ORNL/TM-2012/320, Oak Ridge, TN, September 2012

Liquid Fluoride Thorium Reactor (Flibe Energy, United States of America)

1. Introduction

The liquid-fluoride thorium reactor (LFTR) design by Flibe Energy is a graphite-moderated, thermal-spectrum reactor with solutions of liquid fluoride salts containing both fissile and fertile materials. Thermal power generated from nuclear fission would drive electrical generation in a closed-cycle gas turbine power conversion system. The objective is to produce electricity at low cost by efficiently consuming thorium.

Mixtures of fluoride salts raised to a sufficient temperature to allow them to liquefy form an ideal medium in which nuclear fission reactions can take place. The ionically-bonded nature of the salts prevents radiation damage to the mixture and allows for operation at high temperature yet at essentially ambient pressure.

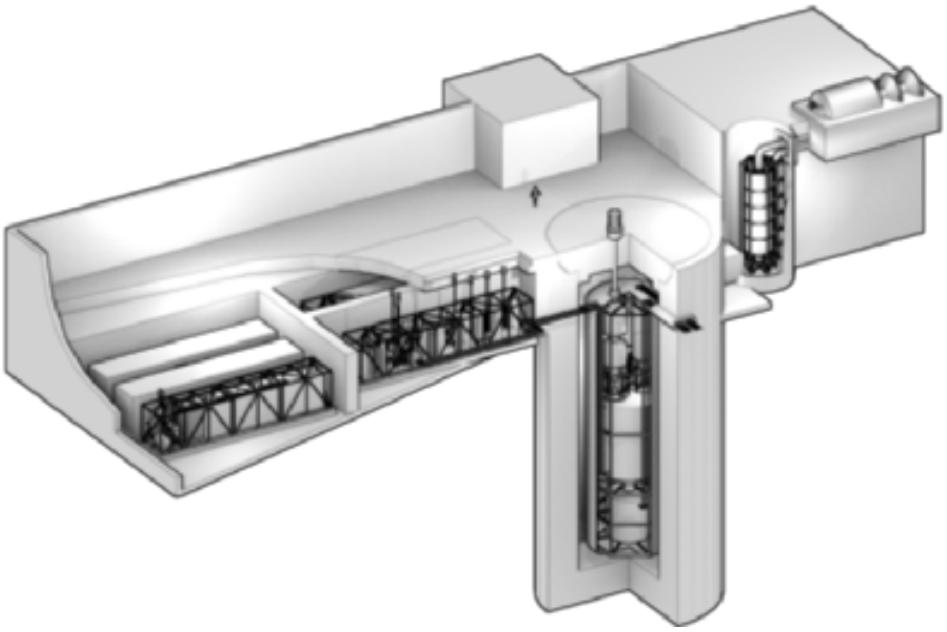

Figure 1: View of the liquid-fluoride thorium reactor (LFTR) plant design (Reproduced courtesy of Flibe Energy)

The high operational temperatures of the fluoride salts (500-700°C) make them excellent candidates for coupling to a closed-cycle gas turbine power conversion system (PCS). The supercritical carbon dioxide gas turbine employing the recompression cycle is proposed and can generate electricity at high efficiencies (approximately 45%).

The LFTR design has a two-region core (feed / breed) and utilizes a closed fuel cycle based on thorium. The reactor vessel incorporates two plena with a central active core region and the outer blanket area, both filled with fluoride salt. The thorium-232 in the blanket region is ultimately converted to uranium-233 through neutron capture and beta decay. The chemical processing system is used to separate and re-introduce the fertile and fissile material to the two fluoride fuel-salt streams respectively. Utilizing thorium fuel in a thermal neutron spectrum, the reactor is able to extract almost all the energy content thus assuring practically unlimited thorium resources and the associated insignificant basic fuel costs.

2. Target Application

Develop a power-generating nuclear reactor that will produce electrical energy at low cost by efficiently consuming thorium.

3. Development Milestones

2015	Completion of EPRI-funded study of LFTR design

4. General Design Description

Design Philosophy

The objective of the liquid-fluoride thorium reactor (LFTR) design proposed by Flibe Energy [1] is to develop a nuclear power plant that will produce electrical energy at low cost. By utilizing thorium fuel in a thermal neutron spectrum, the reactor is able to utilize the energy content of the thorium at a very high efficiency, approaching 100%, at which point the Earth's thorium resources practically becomes unlimited. Some of the main principles followed in the design are (i) inherently safe, with a no meltdown and non-pressurized core; (ii) simplicity, to have an intrinsically stable and self-regulating design; (iii) fuel efficient, and (iv) have the potential to produce far less waste. The general plant layout is shown in *Figure 1*.

Nuclear Steam Supply System

The nuclear heat supply and power conversion system is shown in *Figure 2*. The individual systems are described in more details below followed by other system design descriptions.

Reactor Core

The reactor vessel functions to hold fuel salt, blanket salt, and moderator material together in such a way so as to maintain a critical configuration at the temperatures and thermal power levels required. In addition, it incorporates reactivity control mechanisms both active and passive. The fuel and blanket salts are kept separated in two plena integrated into a single structure within the reactor vessel. Fuel salts are directed into the appropriate channels as it is circulated through the reactor.

The reactor vessel design incorporates several safety functions. In many accident events, a freeze valve, which form part of the vessel and primary loop system, melts and allows fuel salt to drain from the primary loop and the reactor vessel into the drain tank. The separation of the fuel salt from the solid graphite moderator retained in the reactor vessel, assures a subcritical configuration can be established in the drain tank.

The internal graphite structures needs to be replaceable since it subjected to a fast and thermal neutron flux that is greatly in excess of that which will be experienced by the metallic reactor vessel itself, and the replacement of these graphite structures will enable the reactor vessel to continue to operate and serve its function.

Fuel Characteristics

Thorium fuel is introduced as a tetrafluoride into the blanket salt mixture of the reactor. The blanket salt surrounds the active "core" region of the reactor and intentionally absorbs neutrons in the thorium, which leads to the transmutation of the thorium-232 via nuclear beta decay, first to protactinium-233 and later to uranium-233. Both the protactinium and the uranium are chemically removed from the blanket salt mixture and introduced into the fuel salt mixture in the reactor to fission. The fission products are later chemically removed from the fuel salt and in some cases separated and purified before final disposition.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	Flibe Energy
Country of origin	United States
Reactor type	Molten Salt Reactor
Electrical capacity (MWe)	250
Thermal capacity (MWth)	600
Expected capacity factor (%)	90+
Design life (years)	Undetermined
Plant footprint (m^2)	To be determined
Coolant/moderator	Graphite moderator, LiF-BeF2-UF4 fuel salt
Primary circulation	Forced circulation
System pressure (MPa)	Ambient
Core inlet/exit temperatures (^oC)	500 / 650
Main reactivity control mechanism	Negative temperature coefficient; control rod insertion
RPV dimensions and weight	To be determined
Configuration of reactor coolant system	Loop
Power conversion process	Supercritical CO2 gas turbine
Cogeneration / Process Heat Capabilities (High / Low	Example: Yes, possible with different configuration
Passive Safety Features:	Freeze plug releasing to drain tank; large negative
Active Safety Features:	Motor-driven control rods for shutdown
Moderator type/assembly array	Graphite prisms, triangular pitch
Moderator assembly active length (m)	To be determined
Number of moderator assemblies	To be determined
Fuel enrichment (%)	Not applicable, uses uranium-233 fuel derived from Th
Fuel cycle (months)	Continuous refueling from U-233 produced in blanket
Approach to engineered safety systems	Passive safety implemented throughout
Number of safety trains	Not applicable
Emergency Safety Systems	Not applicable
Residual Heat Removal System	Drain tank in thermal communication with environment
Refuelling outage (days)	Not applicable
Distinguishing features	Complete consumption of thorium resource for energy
Modules per plant	4-6
Target construction duration (months)	To be determined
Seismic design	To be determined
Core damage frequency (per reactor-year)	Not applicable
Design Status	Concept

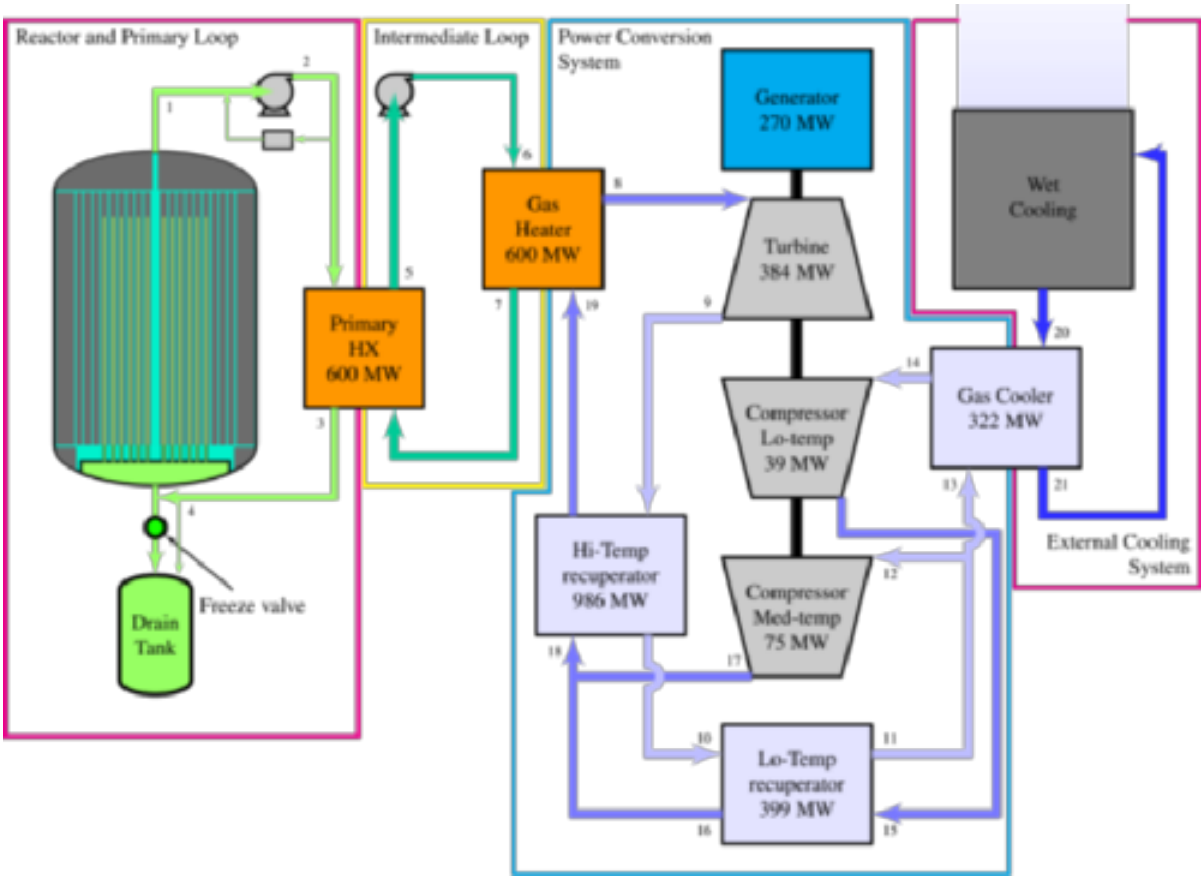

Figure 2: Reactor and primary loop, intermediate loop, power conversion system, and external cooling system simplified flow diagram (Reproduced courtesy of Flibe Energy)

Primary Loop

The function of the primary loop is to direct fuel salt through the primary heat exchanger (PHX) in normal operation, where the fuel salt transfer its heat to the coolant salt. The primary pump provides the necessary forced circulation. The primary loop system begins and ends with its connection to the reactor vessel and includes the primary pump, the PHX, the bubble injection system, and the fuel salt drain tank and its associated external cooling system.

Intermediate Loop

The intermediate loop transfer heat from the primary loop to the PCS The intermediate loop system includes the PHX, the coolant salt pump, the salt side of the gas heater (or intermediate heat exchanger, IHX), the coolant salt drain tanks, and the pressure relief (blowout) valves.

The intermediate loop also isolates the primary loop from the high pressures of the PCS using pressure relief valves. The isolation is an important safety function. In case of a failure in the high pressure PCS it will prevent the transmittal of high pressure back through the coolant salt to the primary loop. The primary loop is not designed for high pressures and without isolation a break in the PCS could cause component rupture and potentially disperse radioactivity into the containment.

In the event of a failure in the gas heater and the pressurization of the intermediate loop, the pressure relief valves allow coolant salt to leave the loop. This deprives the primary loop of cooling capability and will lead the melting of the freeze valve in the primary loop and the

drain of the primary loop fluid contents into the fuel salt drain tank (also see passive shutdown and heat removal later).

The intermediate loop also serves another practical purpose. Since cooling fuel salt with a coolant salt is more compact and effective than cooling fuel salt directly with a gas, the PHX is much smaller and operates at low pressures. It also reduces the fuel salt inventory of the primary loop which reduces the amount of fissile material needed for a given power rating.

Power Conversion System
The function of the PCS is to convert the maximum amount of enthalpy contained in the heated working fluid into shaft work and to reject the remaining enthalpy to the environment in an acceptable manner. The supercritical carbon dioxide gas turbine employing the recompression cycle appears to be the best candidate for coupling to the reactor.

The PCS includes four heat exchangers: the gas side of the gas heater, the gas cooler, and the high-temperature and low-temperature recuperators. It also includes the main turbine, the main compressor, the recompressor, and the electrical generator. The PCS interfaces with the intermediate loop through the gas heater, and interfaces with the external cooling system through the gas cooler.

The PCS and carbon dioxide working fluid in the cycle provides a final barrier to tritium release into the environment. Tritium generation is an inevitable consequence of using lithium and beryllium in the salt mixture and thus the PCS also includes a tritium removal system.

External Cooling System
The function of the external cooling system is to reject the heat that was not converted to shaft power in the PCS to the environment in an acceptable manner. The design shall also prevent the transmission of tritium to the outside environment.

Fuel Handling System
New fissile material can be easily added or removed without resorting to changing the chemical nature of the solvent, and this allows overall reactivity to be held very close to the minimum amount needed to achieve criticality. The fuel salt control is done by the chemical processing system.

Chemical Processing System
The main function of the chemical processing system is to remove uranium and protactinium from the blanket salt and to return uranium to the fuel salt. Its secondary function is to remove fission products from the fuel salt and to further process them into acceptable forms.

The safety-related functions of the chemical processing system mainly involve the safe handling of highly radioactive materials. Drain tanks and cooling systems must be provided for each reaction vessel at each stage of processing through the system. Unfortunately highly chemically reactive gaseous fluorine and hydrogen is needed, to be created just as they are needed from an electrolytic cell using anhydrous hydrogen fluoride as the feed.

Various fission products disperse into the fluid streams of the chemical processing system and some must be handled differently than others. A class of fission products including selenium and tellurium will migrate with gaseous hydrogen and hydrogen fluoride and are handled in a potassium hydroxide neutralization cleanup system. Other fission products are removed from the fuel salt in a reductive extraction column and will exist in a metallic state in bismuth. The high chemical potential of a metal form means that these fission products will need to be

oxidized and placed in a disposal form before shipment from the site. The small amounts of material produced means that these disposal plans will not constitute a major issue with reactor operations.

Substantial development work will be needed to prepare for long-term operation of the chemical processing system.

Offgas Handling System

The function of the offgas handling system is to provide a sufficient holdup volume for xenon and krypton generated in the fission reaction, allowing all of their radionuclides to decay to other forms with the exception of krypton-85, which has a 10-year half-life.

Xenon and krypton (and to some degree tritium) are the most mobile radioactive terms in the reactor system. Tritium is subject to chemical reactions but xenon and krypton are not. Fortunately, with the exception of krypton-85, all of their radionuclides are short-lived and a holdup of roughly thirty days is sufficient to allow them to decay to non-mobile forms like caesium, rubidium, strontium, and barium.

The offgas handling system utilizes the fuel salt decay tank as a primary storage volume, allowing the initial and most intense stages of decay to take place there. The passive cooling system of the fuel salt decay tank is utilized to cool the noble gases, providing a continuous test of the efficacy of this crucial subsystem. After initial cooling in the decay tank, gaseous xenon and krypton in a stream of helium pass into a long piping arrangement filled with charcoal (that adsorbs these gases) and cooled by water that provides sufficient holdup volume over time.

After all radioisotopes of xenon have decayed away, the remaining gas stream is cryogenically cooled to separate stable xenon from helium and krypton. Xenon is bottled and could be sold at this stage. Krypton is also bottled and stored because of the continuing slow decay of krypton-85. Helium is recycled and returned to the gas handling system.

Reactivity Control

The reactor vessel accommodates passive and active control rod systems which also have important safety functions. The blanket salt held within the reactor vessel is a strong neutron absorber, and a blanket salt leak from the reactor vessel could lead to the reduction in the blanket salt inventory contained in the reactor vessel, increasing reactivity by removing a neutron-absorbing medium. To compensate for this introduction of positive reactivity, a series of control rods that float in the blanket salt and are thus held outside of the core could be used. An accidental drain of the blanket salt would remove the buoyancy effect of these rods, allowing them to slide down into the core and add negative reactivity to replace and overcome the negative reactivity lost from by the drain of the blanket fluid. These rods would be designed to enter the core passively, without any operator action, in the event of blanket loss. But it is anticipated that there would also be an active drive system present that could drive these rods into the core intentionally in order to have a shutdown effect on the reactor. It would not be possible to start the reactor unless these rods were fully withdrawn from the core due to their strong negative reactivity.

An active set of control rods, of a more conventional design, would also be present in the reactor vessel and would serve a safety function, allowing the operator to control the reactivity level of the reactor. These rods, which would comprise a smaller and less potent source of negative reactivity, would be clustered near the center of the core and provide finer control over reactivity levels. Other possible designs are also considered.

Reactor Pressure Vessel
The reactor vessel shall be constructed from a material that is suitable for accomplishing its functions at the anticipated temperatures, stresses, and neutron fluxes that will exist during operation. Current evidence points to a modified form of Hastelloy-N as the suitable construction material.

At present, it is anticipated that the reactor vessel will incorporate a small heat exchanger exclusively meant for cooling the blanket salt, with the blanket salt flowing throughout the core under the driving force of natural circulation. This heat exchanger has an important safety function in that it cools blanket salt which contains short-lived thorium-233, a significant heating term in the fluid that cannot be chemically removed. If the reactor shut down or if blanket salt chemical processing was terminated for any reason, the reactor vessel would also have to accommodate the heating generated by protactinium decay, but thorium-233 decay would end relatively quickly in this case, since thorium-233 only has a 22-minute half-life.

5. Safety Features

The properties of fluoride salts offer LFTR enhanced safety characteristics. The fluid salt in the core is not pressurized, thus eliminating the fundamental driving force that can release large amounts of radioactivity to the environment. The notion of a "meltdown" leading to reactor failure becomes irrelevant in a reactor designed around the use of liquid fuels. The reactivity of the reactor is self-controlling because any increase in the reactor's operating temperature results in decrease in density of the fuel salt in the core and a reduction of reactor power, thus inherently stabilizing the reactor without the need for human intervention or backup systems.

The reactor is designed with a simple salt plug drain in the bottom of the core vessel to completely shut down the core through gravity. If the reactor should lose power or need to be powered down for any reason, the salt plug is simply allowed to melt and the fluid salt to drain into a passively cooled containment vessel(s) where decay heat is readily removed. This simple feature prevents accidents or radiation releases due to lack of cooling. It also provides a convenient means to shut down and restart the reactor quickly and easily.

The safety function of several of the systems was already described above. Below a few additional safety features of the LFTR are highlighted or further explained.

Fission Product Retention
The integrity of the reactor vessel plays an important role in minimizing radiation hazards by confining radioactive fluids to the flow channels and volumes defined by the vessel and its internal structures. Most fission products, including all of those of greatest radiological concern, form stable fluoride salts that are retained in the overall mixture under all conditions. Fission products gases, whose removal is important from a performance and safety basis, are easily separated from the fluid mixture and allowed to decay to stability in a separate system.

Passive Shutdown and Heat Removal
An important safety function is embedded in the primary loop and is activated when the reactor overheats or loses its coolant flow. A freeze valve is integrated into the primary loop that is maintained frozen by an active coolant system. When this coolant is lost or if the temperature of the system exceeds its cooling capability, the freeze valve fails open and the fuel salt drains out of the primary loop and out of the reactor vessel into the fuel salt drain tank.

The fuel salt drain tank is integrated with a separate cooling system that is passively connected to the outside environment, and provides the necessary cooling for the fuel salt within it.

Fluoride Salt Characteristics

The fluoride salt mixtures in question have high volumetric heat capacity, comparable to water, and do not undergo vigorous chemical reactions with air or water in contrast to many liquid metals.

The components of fluoride salt mixtures have both desirable and undesirable aspects, and the two most important are lithium-7 fluoride and beryllium fluoride. The two natural isotopes of lithium must be separated from one another since lithium-6 (7.5% of natural lithium) is far too absorptive of neutrons to be a suitable component of a reactor fluid. Beryllium fluoride is chemically toxic but has very attractive nuclear and physical properties. The chemical processing and purification of fluoride salt mixtures typically involves using powerful reactants such as gaseous fluorine and hydrogen fluoride which are very toxic and reactive. But the fact that fluoride salt mixtures are processed in a salt form rather than being dissolved into an aqueous solution mitigates issues of accidental criticality considerably, since water is an excellent moderator whereas salts are poor.

Fluoride salts, due to their exceptional chemical stability, have the potential to corrode most structural metal alloys, but some alloys have been developed that hold up very well against any corrosive attack. Invariably these alloys are based on nickel with a variety of other metallic constituents. Fluoride salts moderate neutrons sufficiently on their own to prevent the formation of a truly fast neutron spectrum, but are still insufficiently effective to generate a thermal neutron spectrum. Thus separate moderator materials are necessary for the reactor and graphite has been proven to be very attractive.

Graphite is not wet by the fluoride salts and does not require cladding. If the surface of the graphite is treated so that small pores are closed, most fission product gases can be excluded from the graphite and overall performance will be high. Graphite does experience issues from dimensional distortion over time, but this effect can be quantified and compensated for in reactor design.

6. Plant Arrangement

The plant arrangement has been shown in *Figure 1*. The reactor cavity or silo is below grade and contains the primary circuit

7. Design and Licensing Status

The design is in an early stage of development and licensing activities have not yet been undertaken.

8. Plant Economics

The economic performance of the plant has not yet been modelled in sufficient detail but all indications point towards its performance being strongly competitive with other clean energy sources.

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[1] For more information visit: http://flibe-energy.com/

Mk1 PB-FHR (UC Berkeley, United States)

1. Introduction

The Mark 1 Pebble-Bed Fluoride-Salt-Cooled High-Temperature-Reactor (Mk1 PB-FHR) is a small, modular graphite-moderated reactor. FHRs are differentiated from other reactor technologies because they use high temperature, coated particle fuels, and are cooled by the fluoride salt flibe ($^{7}Li_2BeF_4$). The Mk1 PB-FHR design described here is the first FHR design to propose driving a nuclear air-Brayton combined cycle (NACC) for base-load electricity generation. *Figure 1* shows the flow schematic for the Mk1 PB-FHR.

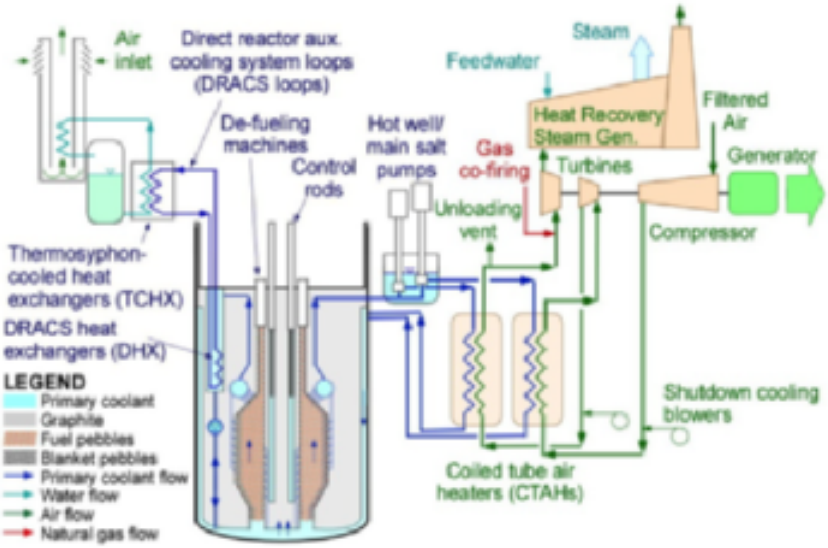


Figure 1: Mk1 PB-FHR flow schematic (Reproduced courtesy of UC Berkeley)

2. Target Application

The Mk1 PB-FHR is designed to produce 100 MWe of base-load electricity when operated with only nuclear heat, and to increase this power output to 242 MWe using gas co-firing for peak electricity generation. This provides a new value proposition for nuclear power to earn additional revenues by providing flexible grid support services to handle the ever increasing demand for dispatchable peak power. This is in addition to traditional base-load electrical power generation.

3. Development Milestones

To be developed

4. General Design Description

Design Philosophy

The Mk1 PB-FHR is designed with advanced passive safety features and intrinsic fuel and coolant properties which make the consequences of severe accidents commonly studied for light water reactors much easier to manage. Passive safety mechanisms include natural circulation decay heat removal activated by a passive check valve in accident conditions and buoyant control rods for emergency shutdown without operator intervention.

Fluoride salt coolants have uniquely high volumetric heat capacity, low chemical reactivity with air and water, very low volatility at high temperature, effective natural circulation heat transfer, and high retention of most fission products. These characteristics are in addition to reasonably low neutron capture probability, and good neutron moderation capability.

The Mk1 design implements several innovative features: The Mk1 PB-FHR does not use an intermediate coolant loop and instead directly heats the power conversion fluid. It eliminates the conventional reactor guard vessel used in sodium fast reactors and instead uses a refractory reactor cavity liner system. Furthermore, all components for the Mk1 design are rail-transportable.

Reactor Core

Figure 2 shows a cross section of the Mk1 reactor vessel, internals, and core. The core design incorporates an annular pebble-bed core geometry composed of a homogeneous mix of fuel pebbles adjacent to the center graphite reflector, with a layer of inert graphite reflector pebbles on the outside that reduces the fast-neutron fluence to the outer fixed radial graphite reflector sufficiently for it to last the life of the plant.

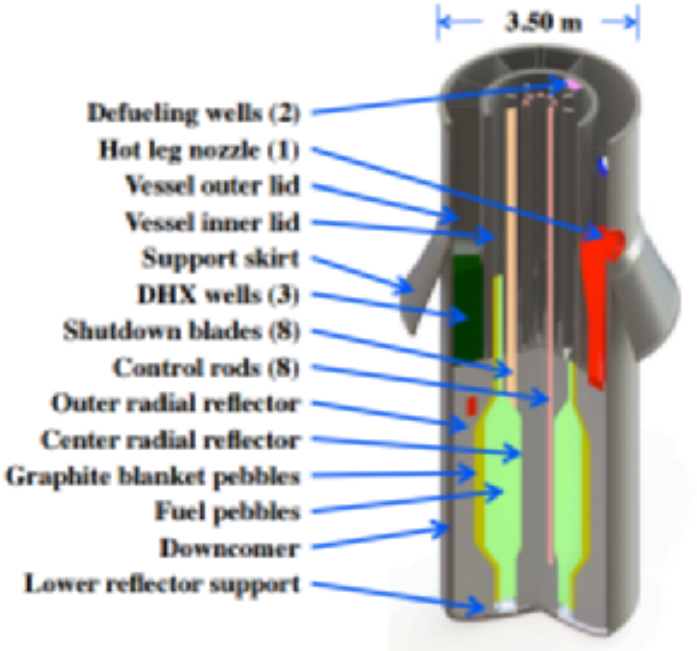


Figure 2: The Mk1 PB-FHR reactor vessel (Reproduced courtesy of UC Berkeley)

The center reflector provides 8 channels for insertion of buoyant control rods, and it also provides flow channels for radial injection of coolant into the pebble core, to provide a combined radial and axial flow distribution that increases the effectiveness of heat transfer from the fuel and results in lower average fuel temperature. The center graphite reflector internals need to be replaced periodically due to radiation damage. Its size is much shorter and smaller than center reflectors proposed for helium-cooled pebble-bed reactors, making it simpler to design for replacement and for seismic qualification.

Power Conversion Unit

The 236-MWth Mk1 PB-FHR uses the NACC power conversion system. It uses a General Electric (GE) 7FB gas turbine (GT), modified to introduce external heating and one stage of reheat, in a combined-cycle configuration to produce 100 MWe under base-load operation, and with natural-gas co-firing to rapidly boost the net power output to 242 MWe to provide peaking power. The Power Conversion Unit consists of the reactor core, two coiled tube air

heaters (CTAHs) to transfer heat from the main salt to pressurized air, a heat recovery steam generator system (HRSG), steam condenser, and the GT. During normal operation, the primary coolant relies on forced circulation.

Fuel Characteristics

Figure 3 shows the 3-cm diameter spherical pebble fuel elements. The coated uranium particles are packed in an annular fuel zone around a low-density graphite core. One Mk1 pebble contains 1.5 g of uranium enriched in U-235 to 19.9% and encapsulated inside 4,730 coated particles. The very low circulating power for the coolant in salt-cooled reactors, compared to helium-cooled reactors, makes it practical to use smaller pebbles. This small-pebble design doubles the pebble surface area per unit volume and halves the thermal diffusion length, enabling a substantial increase in power density while maintaining relatively low peak fuel particle temperature. Four Mk1 pebbles can provide electricity for a full year to an average U.S. household (approximately 11 MWe-hr in 2011). This is equivalent to 8.1 tons of anthracite coal, or 17 tons of lignite coal, if produced in a coal fired plant.

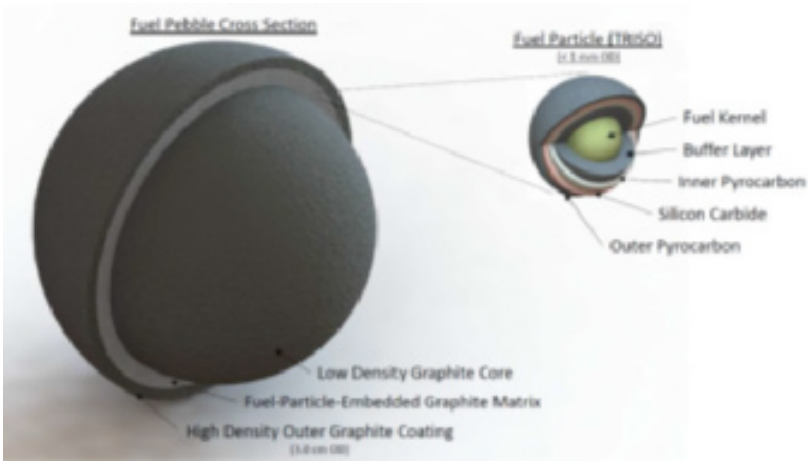


Figure 3: A PB-FHR pebble fuel element (Reproduced courtesy of UC Berkeley)

Fuel Handling System

The design includes pebble injection channels integrated into the inner surface of the reactor core barrel. The pebble injection channels follow the curvature of the bottom of the reactor vessel before discharging pebbles through the lower center reflector support structure. The Pebble Handling and Storage System (PHSS) will function to ensure the sustained fission reaction in the core through the addition and removal of fuel and graphite pebbles. Space has been allocated in the upper core internals and in the reactor building for key PHSS subsystems and components including the core unloading devices, pebble burnup measurement system, and pebble canister transfer system, but detailed design has not yet been completed.

Reactivity Control

For reactivity control, the Mk1 is designed to have negative fuel, moderator, and coolant temperature reactivity feedbacks. The design uses a buoyant control rod system for normal reactivity control, and the system also provides a passive shutdown capability because the buoyant rods will insert if the reactor coolant temperature in the control-rod channel exceeds 615°C, the buoyant stability limit. The design also uses shutdown blades that can insert directly into the pebble bed for reserve shut down. In the event that electrical power is interrupted to the drive mechanisms for the motors of the control rod and shutdown blade cable drums, they will insert and shut the reactor down.

MAJOR TECHNICAL PARAMETERS	
Parameter	**Value**
Technology developer	University of California, Berkeley
Country of origin	United States of America
Reactor type	Fluoride-Salt-Cooled High Temperature Reactor (FHR)
Electrical capacity (MW(e))	100 MW_e
Thermal capacity (MW(th))	236 MW_{th}
Expected capacity factor (%)	>90
Design life (years)	60
Plant footprint (MW_e/acre)	16.1
Coolant	Li_2BeF_4 (flibe)
Moderator	Graphite
Primary circulation	Forced circulation
System pressure (MPa)	0.3
Core inlet/exit temperatures (^oC)	600 / 700
Main reactivity control mechanism	Negative temperature coefficient; control rod insertion
RPV height (m)	12
RPV diameter (m)	3.5
RPV weight (metric tonnes)	62
Configuration of reactor coolant system	Pool type
Power conversion process	Nuclear air-Brayton combined cycle (NACC)
Cogeneration / Process Heat Capabilities (High / Low T)	Yes, possible with different configuration
Passive Safety Features:	Large negative temperature coefficients, large heat capacity
Active Safety Features:	Yes, control rod SCRAM; Turbine trip
Fuel type/assembly array	TRISO particles in graphite pebble matrix / Pebble bed
Active height of fuel region (m)	2.5
Number of fuel pebbles in core and defueling chute	470000
Fuel enrichment (%)	19.9
Fuel burnup (MWd/kgHM)	180
Fuel core residence time (months)	2.1, average of 8 passes to achieve full burn up
Approach to engineered safety systems	Passive
Emergency Safety Systems	Control rod drive mechanism, shutdown blades
Residual Heat Removal System	Direct reactor auxiliary cooling system (DRACS)
Refuelling outage (days)	0 days (online refuelling)
Distinguishing features	Large fuel and coolant thermal margin to damage, high temperature operation, NACC with peaking, passive safety
Modules per plant	12
Target construction duration (months)	24
Design Status	Pre-Conceptual Design

Reactor Pressure Vessel and Internals

To enable near-term deployment, the Mk1 design uses a core barrel and other core internal structures fabricated from the same metallic material as the reactor vessel and main salt piping. The outer radial reflector blocks are aligned and held against the metallic core barrel using a system of axial alignment ribs and radial retaining rings quite similar to designs

originally developed for the Molten Salt Breeder Reactor (MSBR) project. The use of metallic core internal structures, rather than advanced ceramic composites, simplifies fabrication and licensing for the Mk1 design.

Steam Generator

The Heat Recovery Steam Generator System (HRSG) and steam condenser need to be sized for full power operation at co-firing conditions. The large HRSG inlet temperature variation between baseload and co-fired operation modes introduces certain caveats to the steam cycle design. With the expected frequent power ramping of the GT and dissimilar ramping rates compared to the steam turbines/HRSG, special design considerations are needed such as opening steam turbine inlet valves and or allowing some bypass flows.

5. Safety Features

Mk1 PB-FHR reactor protection functions are performed by a combination of passive mechanisms, control signals and actuators from the safety-grade digital reactor protection system (RPS), and from manual operator actions.

Engineered Safety System Approach and Configuration

For reactivity control, the Mk1 has a combination of intrinsic features and passive systems. It has negative fuel, moderator, and coolant temperature reactivity feedbacks. The reduced fuel temperature in the PB-FHR provides improved response to hypothetical ATWS accidents. The negative fuel temperature reactivity feedback in FHRs is significantly larger than the coolant temperature reactivity feedback, because the coolant does not boil—the boiling temperature of flibe is 1430°C—as in light water reactors (LWRs), and larger than the graphite moderator temperature reactivity feedback. Under the beyond design basis ATWS accident where reactor scram does not occur upon loss of flow or loss of heat sink, the FHR coolant equilibrates to a temperature close to the original fuel temperature. Simplified analysis for the Mk1 design indicates that this equilibrium ATWS temperature will be below 800°C.

Decay heat removal system

In the PB-FHR core, the emergency heat removal safety function is also controlled by passive mechanisms. The PB-FHR design concept employs a passive check valve to activate natural-circulation-driven heat transport to a set of three Direct Reactor Auxiliary Cooling System (DRACS) loops and ultimately to Thermosyphon-cooled Heat Exchangers (TCHXs) upon Loss of Flow Condition (LOFC). Heat removal from the TCHXs is regulated by fail-open valves that supply water to the thermosiphons integrated into these heat exchangers. The valves are held closed during normal operation, and can also be closed to control over-cooling during prolonged reactor shutdown. In addition to the passive emergency decay heat removal provided by the DRACS, the PB-FHR power conversion system and the normal shutdown cooling system provide heat removal capability and defense in depth in assuring adequate core heat removal.

Emergency core cooling system

Coolant inventory control is provided by fully passive mechanisms that require no RPS or manual operator actions. The primary salt fulfils dual roles during design basis events, by providing natural-circulation heat removal and preventing chemical attack to fuel pebbles from exposure to air. The PB-FHR utilizes a pool-type reactor configuration, similar to the design adapted for many sodium fast reactors. For BDBEs where the vessel leaks or ruptures, the Mk1 refractory cavity liner insulation system controls the level change in the vessel and prevents uncovering of fuel.

Containment system
The Mk1 design introduces another novel feature, a "gas gap" system, to make it physically impossible to transmit excessive pressures to the reactor vessel and reactor cavity/containment from potential tube or manifold pipe ruptures in a CTAH. The gas gap is created adjacent to the containment penetrations for the hot and cold legs.

For the Mk1 PB-FHR, water pools are used inside the shield building to provide water to thermosyphon-cooled heat exchangers (TCHXs) in the DRACS modules, as well as to the reactor cavity liner cooling system. Because these water pools also provide a source of water for evaporative cooling under beyond-design-basis event (BDBE) conditions, they are provided with a secondary confinement following the "tank-within-tank" design principle.

6. Plant safety and Operational Performances

Due to the high thermal efficiency of the NACC system, the steam-bottoming condenser requires only 40% of the cooling water supply that is required for a conventional LWR, for each MWh of base-load generation. As with conventional natural-gas combined cycle (NGCC) plants, this makes the efficiency penalty of using dry cooling with air-cooled condensers much smaller, enabling economic operation in regions where water is scarce. The advantage of the NACC system arises from additional revenues earned by providing flexible grid support services because under base-load operation NACC power conversion has lower fuel costs than NGCC, and under peaking operation has higher efficiency in converting natural gas to electricity than NGCC, NACC plants will always dispatch before conventional NGCC plants.

7. Instrumentation and Control systems

The digital control system is designed so that neither its actions nor its failure to act would have any deleterious impact on the ability of the PB-FHR to respond safely to design basis events. The quality requirements for the control system then arise from the economic incentives to maximize system performance and to preserve the invested capital, thus high-quality commercial-grade equipment is anticipated to be used.

Except during startup and low-power conditions, the PB-FHR operates with constant core inlet and outlet temperatures. Load-following capability is made possible by air bypass flow to respond to rapid load-change transients and turbine inlet temperature control (by bypassing air around the CTAHs) for slower transients. Pump speed control is then used to control the core temperature difference, and control rod position is used to control the average temperature. The control system adjusts the pebble loading and unloading schedule to maintain sufficient excess reactivity to accommodate a xenon transient equivalent to a rapid power reduction from 100% to 40%.

The PB-FHR Post-Event Instrumentation (PEI) system will monitor primary system temperatures, as well as other safety related parameters, after design basis events. Data provided by the PEI allows operators to assess the evolving plant state and take appropriate long-term control actions; including assurance that overcooling does not occur.

The PB-FHR control system is designed to allow the reactor to withstand grid transients, including loss-of-load, by venting hot air downstream of the LP CTAH, followed by a controlled reactor power run-back and activation of the shutdown cooling system to remove decay heat.

The reactor control portion of the PB-FHR plant control system will have common sensor elements to the RPS. All communications from the RPS to the control system will be via buffered one-way communication links (i.e., IEEE 7.4.3.2-2003 Annex E type links). The common sensing elements between the reactor safety and control system will include power range neutron flux, primary coolant temperature, primary coolant level and inventory, and primary coolant flow rate.

8. Plant Arrangement

Figure 4 presents a notional 180-acre site arrangement for a 12-unit Mk1 PB-FHR power plant capable of producing 1,200 MWe base load and 2,900 MWe peak power output. Due to the much smaller cooling requirements they do not need to be sited near bodies of water. Population centers tend to be located near bodies of water which means that FHRs can be sited in areas where fewer people want to live. So, rather than attempt to minimize the site footprint, the more important goal is likely to facilitate construction of modules adjacent to operating modules, and to optimize the degree to which some services are shared.

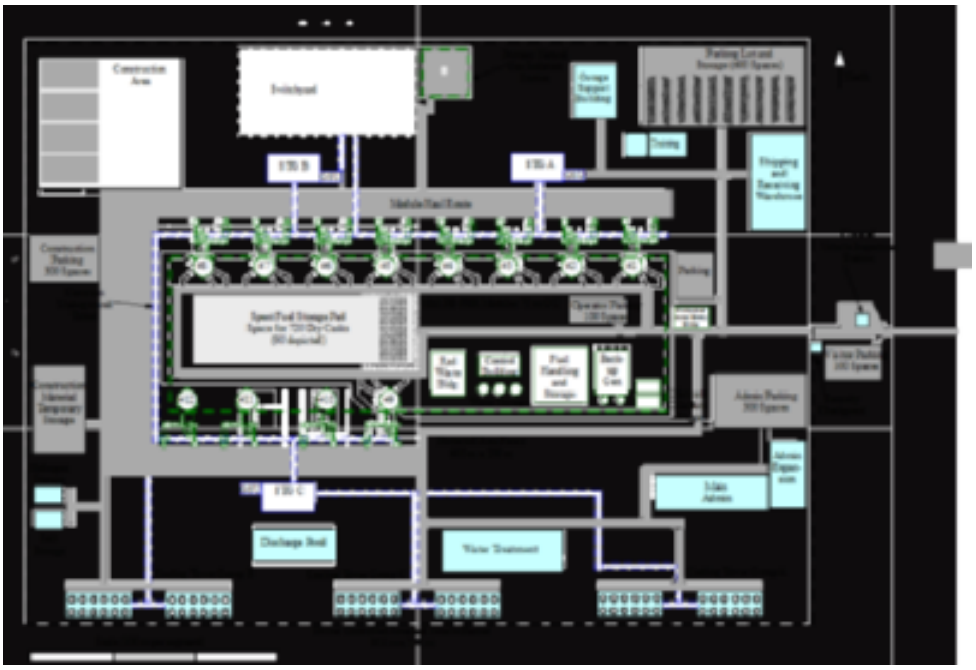


Figure 4: Mk1 site arrangement for a 12-unit, 180-acre PB-FHR plant, capable of producing 1200 MWe base load and 2900 MWe peak power (Reproduced courtesy of UC Berkeley)

Reactor building

The Mk1 reactor building and NACC system arrangements supports a multi-module plant configuration, shown in *Figure 5*, by allowing multiple units to be lined up in a row with a clear boundary between the reactor and its vital areas, versus the balance of plant (BOP). The GT and associated equipment are configured to minimize the air pressure loss and circulating power in the air ducting while maintaining a clear boundary between the reactors and the BOP. This configuration makes it easier to co-locate combined nuclear services on one side of a multi-module plant (training, fresh fuel handling/receipt, spent fuel dry storage, security, access control, multi-module control room, hot-rad/Be shops, etc.), and have BOP combined services on the other side (off-site transmission, process steam loads and/or steam bottoming turbines, cooling towers, etc.).

The Mk1 reactor building is partially embedded below grade, with the reactor deck located slightly above grade, shortening the air duct lengths and the depth of the air-duct vault. The baseline Mk1 reactor building design uses a cylindrical shield building fabricated from steel-plate/concrete composite (SC) modules, quite similar to the Westinghouse AP1000 shield building. The overall height and diameter of the Mk1 shield building are 47.5 m and 24.5 m, respectively, compared to 83 m and 42 m for the 1150-MWe AP1000, so the Mk1 shield building volume is 2.2 times greater than the AP1000, per MWe baseload.

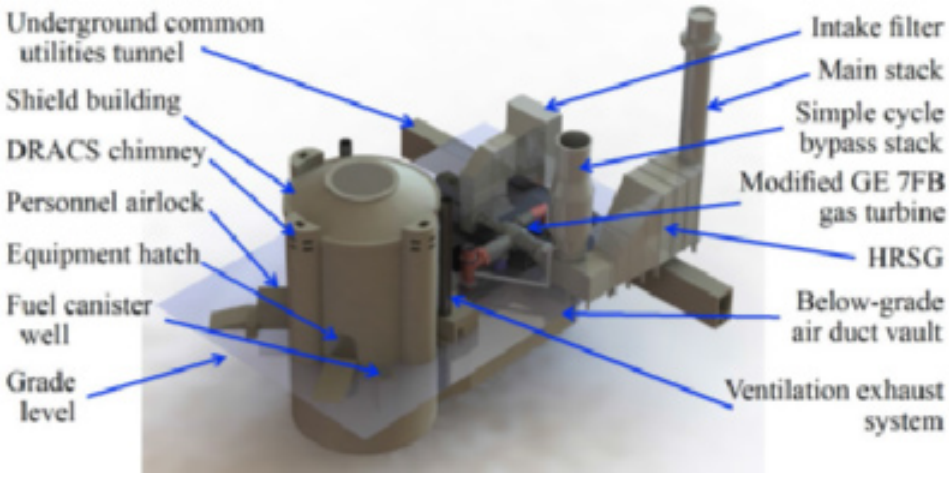


Figure 5: The Mk1 PB-FHR reactor shield building adjacent to the power conversion system (Reproduced courtesy of UC Berkeley)

The cylindrical shield building geometry provides a stiffer, lighter structure than is possible with a rectangular building geometry. The low-leakage containment boundary is provided by the liner of the reactor cavity and its gas space. With the containment boundary located well inside of the reactor building external walls, the primary function of the cylindrical structure is to provide an external events shield. It also acts as an element of the secondary filtered confinement volume used for control of beryllium and radioactive contamination. Because the refueling deck area of the reactor building must be accessed periodically during power operation for fuel transfer work associated with on-line refueling, the refueling deck is designed to provide effective biological shielding.

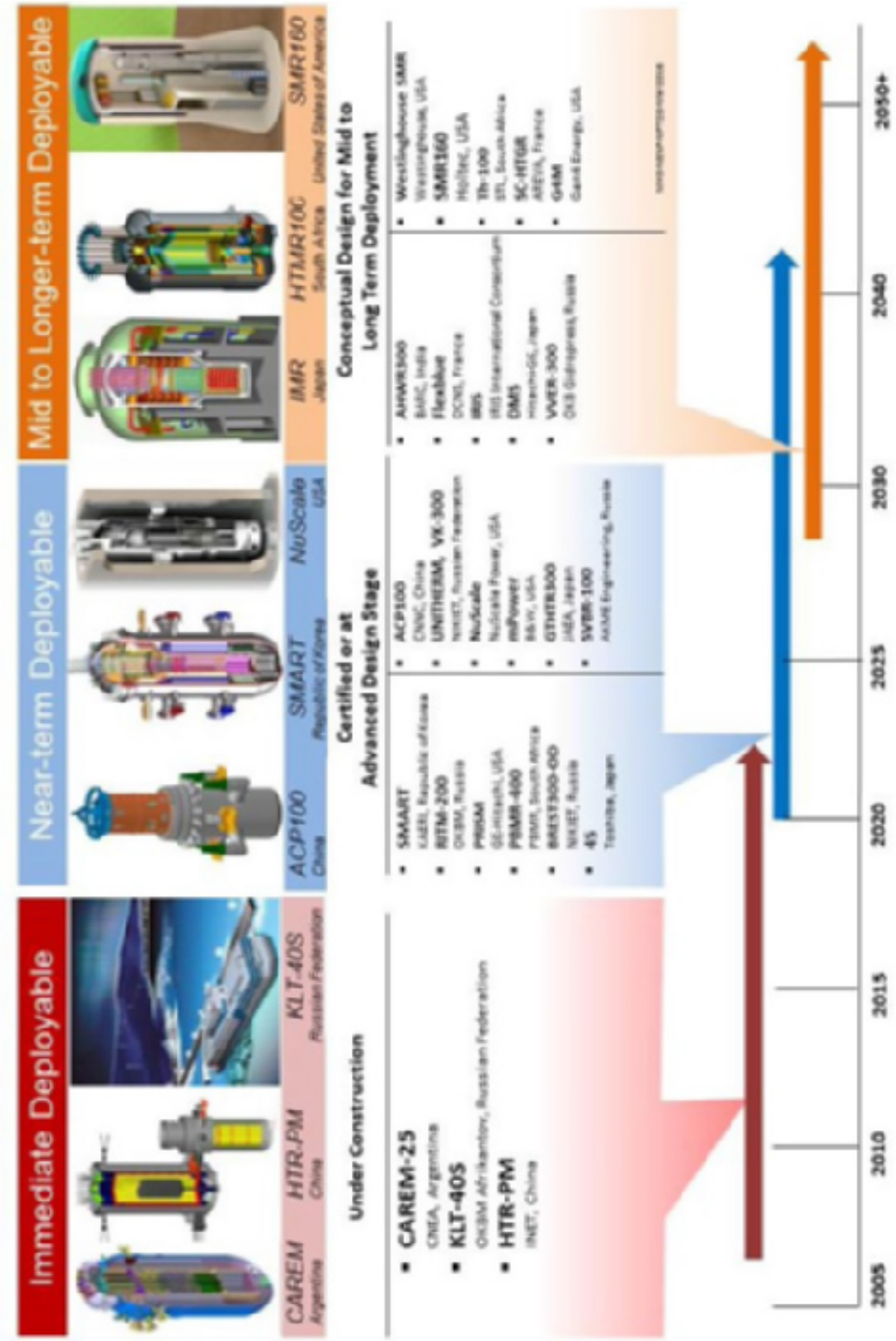
9. Design and Licensing Status

Further work is needed in the definition and design of: plant staff capabilities and size, instrumentation requirements, systems and equipment for operations and maintenance, future plant reliability and availability, and licensing strategies for licensing commercial prototypes in the U.S. as well as internationally.

10. Plant Economics

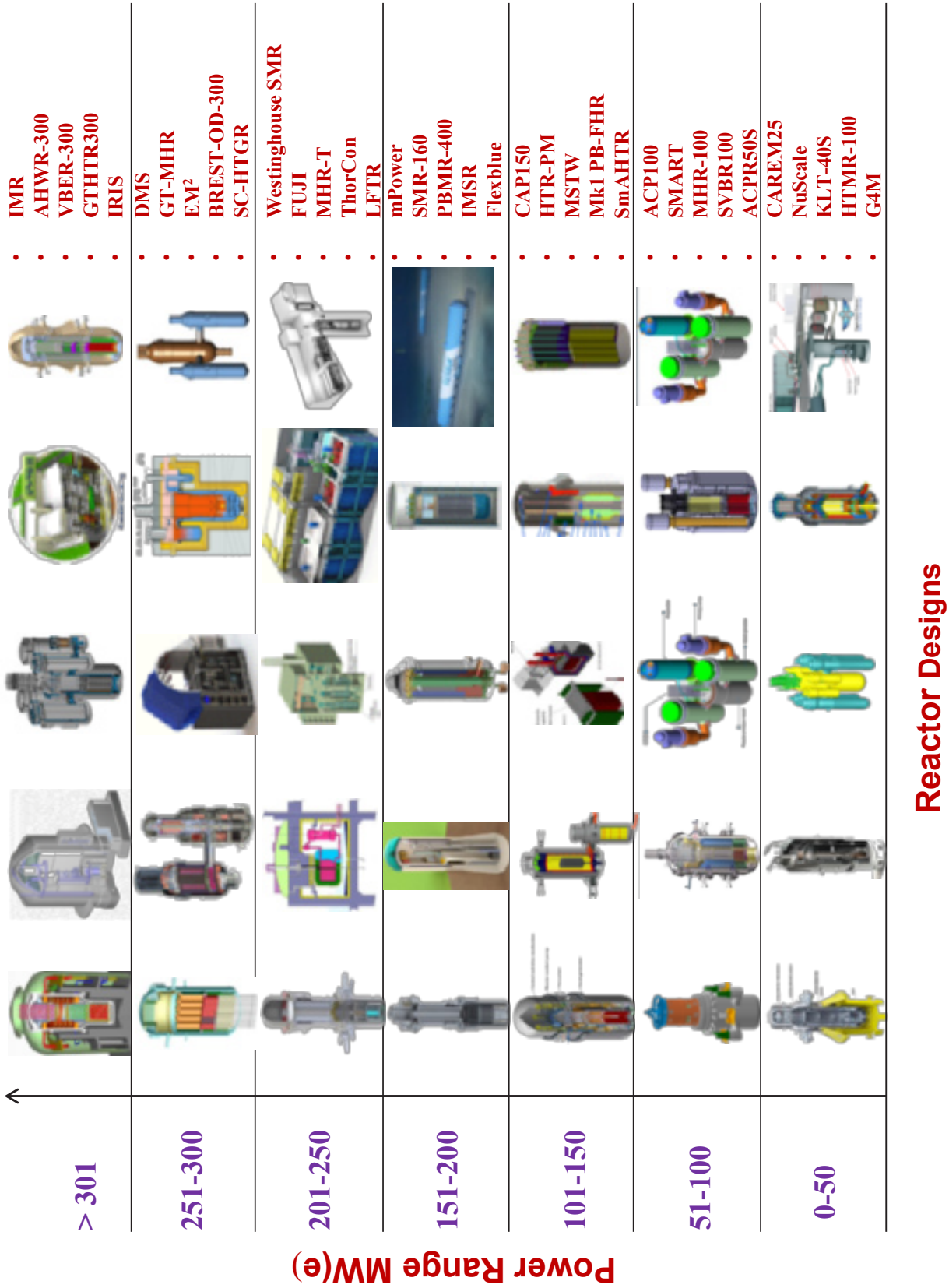
An overall capital cost of $4,500-$5,093/kW base load has been estimated depending on the number of units, ranging from one to twelve, present at the power plant site. Since all infrastructure required to also perform peak load is already included the specific capital is reduced to $1,870-2,133/kW for base load plus peaking. These numbers should be compared to capacity weighted average capital costs of a NGCC and a conventional nuclear power plant, rather than each separately. Similar to capital construction costs, O&M costs drop from $81.05/MWh (¢8.1/kWh) to $39.82/MWh (¢3.98/kWh) as the number of units per site increase from one to twelve.

ANNEX I
SMR Estimated Timeline of Deployment

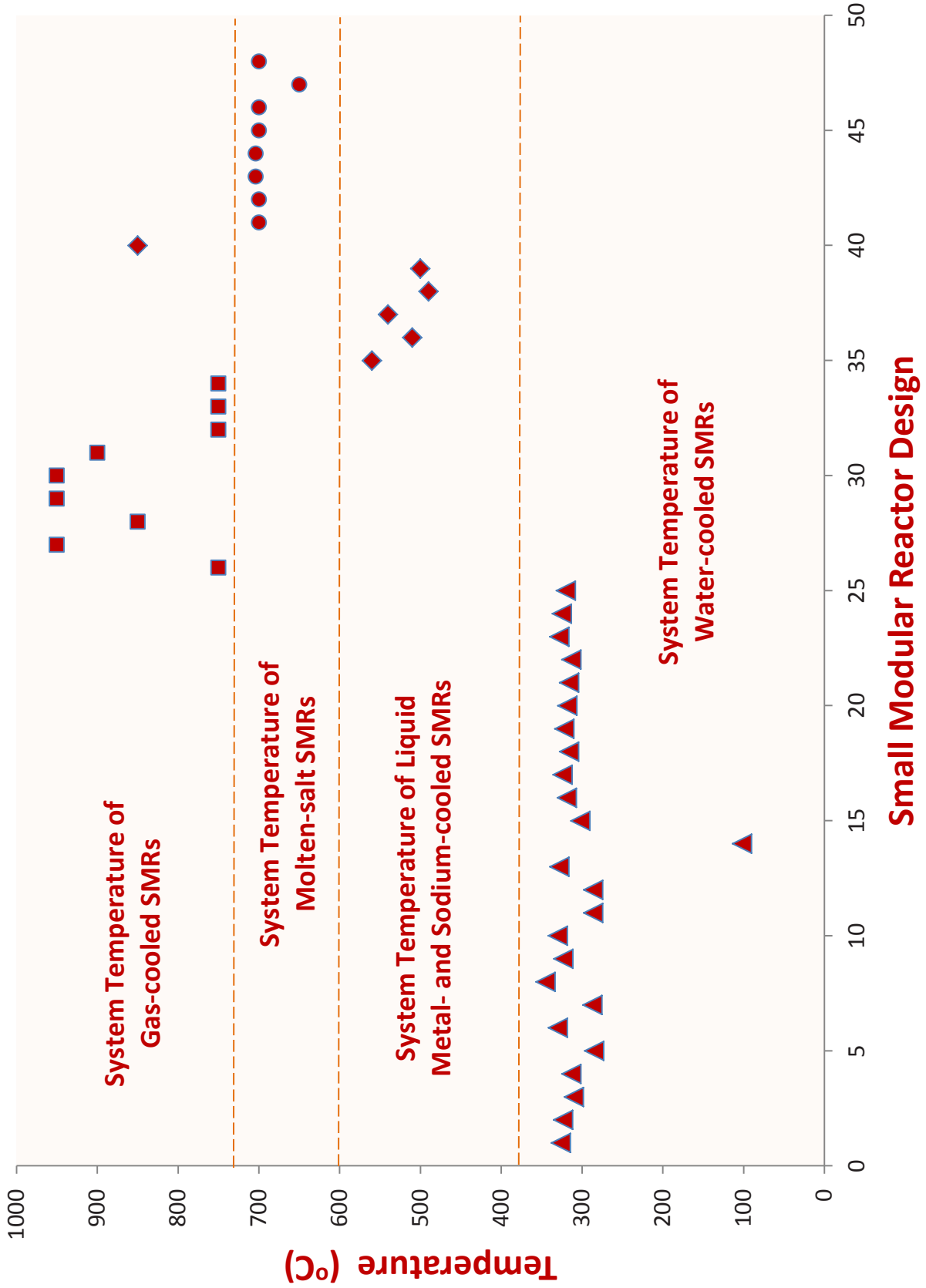


ANNEX II

Summary of SMR Designs Based on Power Range

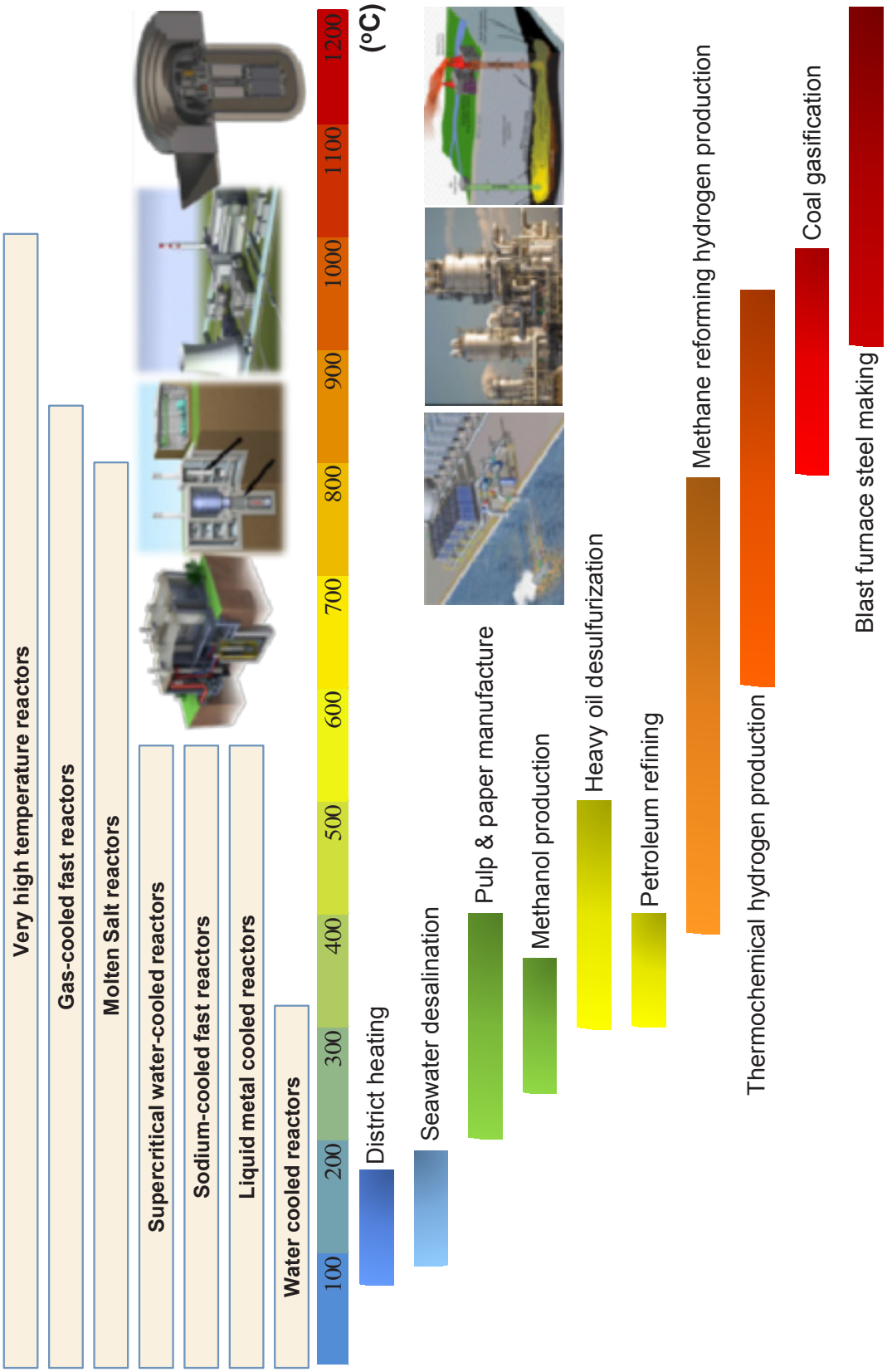


ANNEX III
Summary of SMR Designs Based on Core Exit Temperature



ANNEX IV

Summary of SMR Designs for Non-Electric Applications

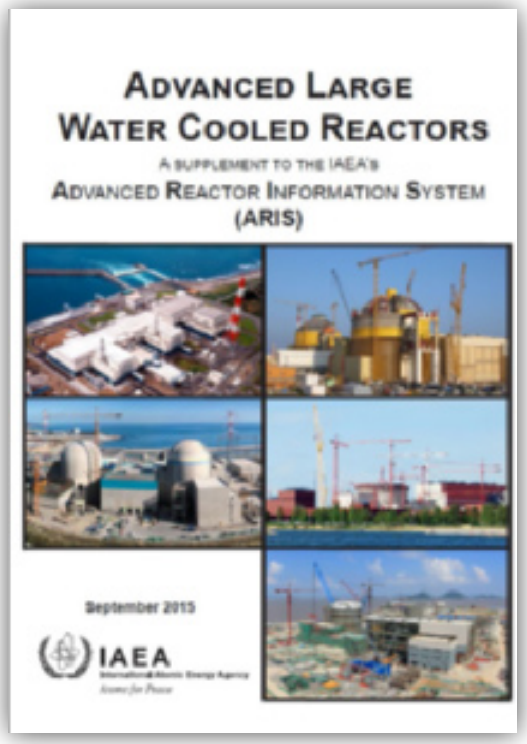


ANNEX V
Bibliography

- 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), NuScale (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).
- Published September 2012

- 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).
- Published October 2013

- Contained status, design description and main features of 31 selected SMR designs;
- Sorted by type/coolant: iPWR, AHWR and HTGR;
- Sorted by Country of Origin;
- Included: CAREM (Argentina), ACP-100 (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), KLT-40S (Russian Federation), UNITHERM (Russian Federation), IRIS (International Consortium), mPower (USA), NuScale (USA), Westinghouse SMR (USA),SMR160 (USA), AHWR300-LEU (India), HTR-PM (China), PBMR (South Africa), GT-MHR (Russian Federation),VVER-300 (Russian Federation), RUTA-70(Russian Federation), ELENA (Russian Federation), DMS (Japan), HTR-PM (China), GTHTR300 (Japan), MHR-T (Russian Federation), MHR-100 (Russian Federation), PBMR-400 (South Africa), HTMR-100 (South Africa), SC-HTGR (USA), Xe-100 (USA)
- Published September 2014



- Contained overview of status and main features of 18 selected large water cooled reactor designs;
- Sorted by Country of Origin/Vendor;
- Included: ACPR-1000 (China), CAP-1400 (China), CPR-1000 (China), HPR1000 (China), APR1400 (Republic of Korea), APWR (Japan), AP1000(Japan), ABWR(Japan), VVER1000(Russian Federation),, VVER1200(Russian Federation),.VVER1500(Russian Federation), IPHWR(India), EPR(France), KERENA(France), ATMEA1(France) , EC6 (Canada), ABWR(USA), ESWR(USA)
- Published September 2015

ANNEX VI
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

For further information:
Nuclear Power Technology Development Section (NPTDS)
Division of Nuclear Power
IAEA Department of Nuclear Energy

International Atomic Energy Agency
Vienna International Centre
PO Box 100
1400 Vienna, Austria

Telephone: +43 1 2600-0
Fax: +43 1 2600-7
Email: Official.Mail@iaea.org
Internet: http://www.iaea.org

Printed by IAEA in Austria
August 2016

16-24301